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Separation Requirements for a Hydrogen Production Plant and High-Temperature Nuclear Reactor

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ABSTRACT

This report presents engineering methods, models, and results to enable an evaluation for locating a hydrogen production facility near a nuclear power plant. Standard probabilistic safety assessment methodologies were used to answer the risk-related questions for a combined nuclear and chemical facility: what can happen? how likely is it to happen? and what are the consequences of it happening? As part of answering these questions, a model was developed suitable to determining the distances for separating hydrogen process and nuclear plant structures. The objective of the model-development and analysis is to answer key safety questions relating to the placement of one or more hydrogen production plants in the vicinity of a high-temperature nuclear reactor. From a thermal-hydraulic standpoint, proximity of the two facilities is beneficial. Safety and regulatory implications, however, force the separation to be increased, perhaps substantially. The likelihood of obtaining a permit to construct and build such as facility in the United States without answering these safety questions is questionable. The quantitative analysis performed for this report offers a scoping mechanism to determine key parameters relating to the development of a nuclear-based hydrogen production facility. The calculations indicate that when the facilities are less than 100 m apart, the core damage frequency is large enough (greater than 1E-6/yr) to become problematic in a risk-informed environment. However, a variety of design modifications (blast-deflection barriers, for example) could significantly reduce risk and should be further explored as design of the hydrogen production facility evolves.

CONTENTS

FIGURES

TABLES

ACRONYMS

BWR	boiling water reactor
GSI	Generic Safety Issue
HTGR	high-temperature fast reactor
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory (formerly Idaho National Engineering and Environmental Laboratory)
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
PSA	probabilistic safety assessment
PWRs	pressurized water reactor
RG	Regulation Guide
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (probabilistic risk and reliability analysis software)
SRP	Standard Review Plan
SSC	structures, systems, and components

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1. ANALYSIS OBJECTIVES

1.1 Background

Producing and storing hydrogen with a nuclear power plant poses unique engineering problems associated with operational safety. As has been done with the current generation of nuclear power plants, in this report we formally evaluate safety using the methods of probabilistic safety assessment (PSA) (Bedford and Cooke 2001). However, unlike conventional analysis, our approach to assessing hydrogen safety in connection with the new generation of nuclear reactors focuses on a multidimensional array of hazards (not simply such a measure as core damage), hazards of interest to both engineers and decision-makers with a stake in the safe and efficient operation of the proposed facility.

In general application, PSA techniques answers three questions:

1. What can happen?
2. How likely is it to happen?
3. What are the consequences of it happening?

This report answers these three questions with regard to evaluating the risks associated with high-temperature hydrogen production. Initially (Section 1), the focus is on the development of a model suitable to assist in determining distances for separating hydrogen process structures and the nuclear plant structures. Then (Section 2), we overview the methods and objectives used to evaluate the decision to locate a hydrogen production facility near a nuclear power plant. Finally (Section 3), we present the results of the detailed analysis.

Our primary objective of the model-development and analysis is to answer key safety questions relating to placement of one or more hydrogen production plants in the vicinity of a high-temperature nuclear reactor. As illustrated in Figure 1, we determine the risk implications as a function of location for the reactor and hydrogen facilities. From a thermal-hydraulic perspective, we would like the two facilities to be quite close. From a safety and regulatory perspective, implications force the distance to be increased, perhaps substantially. Without answering the safety questions posed in this report, the likelihood for obtaining a permit to construct and build such as facility in the United States would be questionable.

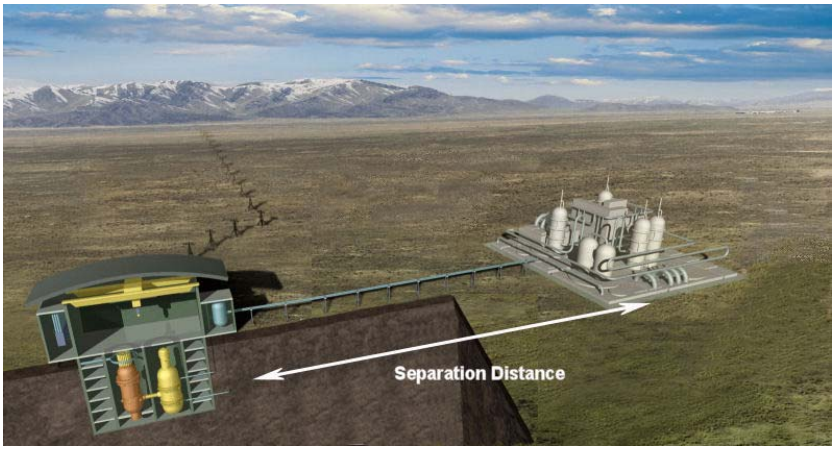

Figure 1. The distance to be evaluated from a safety perspective separating the high-temperature nuclear reactor and the hydrogen production facility.

Our secondary objective of model development and analysis is to provide decision makers tools and insights into potential engineering designs that may mitigate hazards inherent in coupling nuclear and hydrogen production facilities. In addition to estimating risk as the distance separating the facilities is modified, these safety-based tools allow selection of a variety of facility types, e.g., different chemical processes, safety barriers, and inventory levels.

Though currently in substantial use worldwide, hydrogen is not a benign material, as demonstrated by the occurrence of events relating to the production, transportation, and storage of hydrogen (for example, the Stockholm explosion in 1983 (Venetsano et al. 2003) the outcome of which is shown in Figure 2). Generating hydrogen will require understanding its deflagration and detonation potential and the risks associated with using hazardous chemicals during production. Understanding hydrogen risks is critical at such regulatory agencies as the U.S. Nuclear Regulatory Commission (NRC), since hydrogen is used at current generation nuclear power plant facilities. Such plant use of hydrogen has been investigated though such programs as the NRC's Generic Safety Issue 106, *Piping and Use of Highly Combustible Gasses in Vital Areas* (INEL 1994). The safety analysis work proposed in this report will build upon and extend that body of knowledge specific to the high-temperature production of hydrogen in next-generation nuclear power plants.

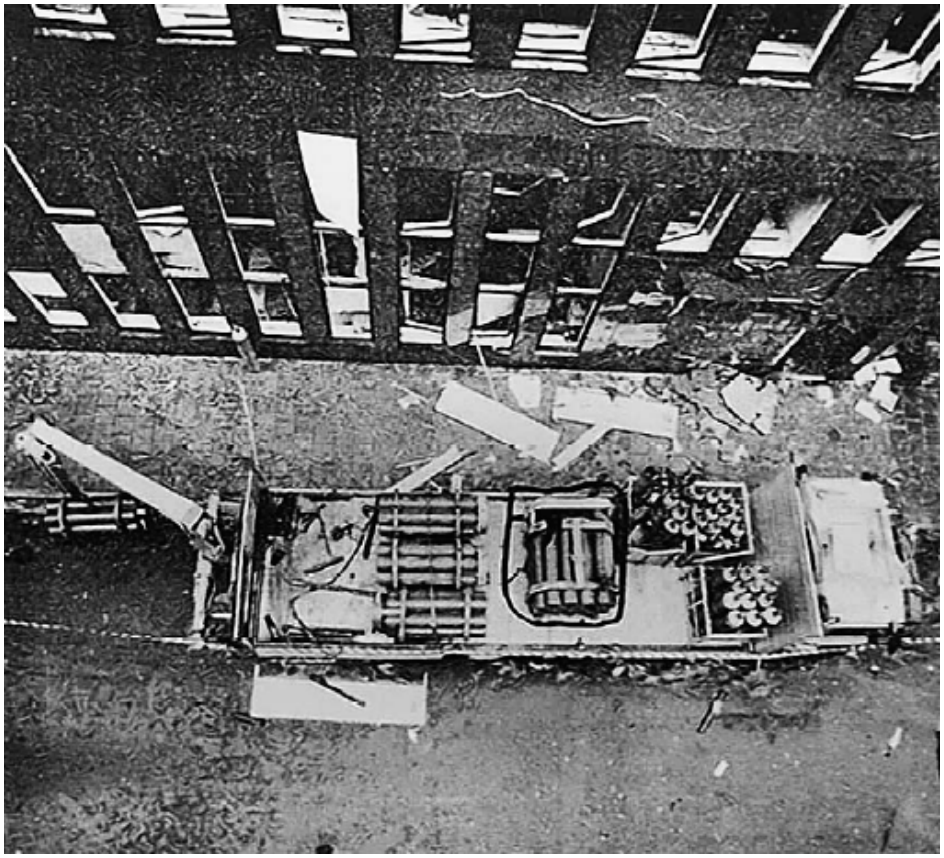

Figure 2. Hydrogen cylinder explosion in Stockholm in 1983 from a total of 13.5 kg of hydrogen (Venetsano et al. 2003).

In addition to hydrogen deflagration and detonation risks, the safety of the associated chemical process must also be addressed. The analysis documented in this report determines the potential for upset conditions to occur to the nuclear facility during hydrogen generation. For example, the risk from acid or

toxic gas clouds from a release must be evaluated. Both hydrogen and chemical-based evaluations and models will be used to provide decision makers insights into the inherent safety of the processing facilities. Thus, these analyses will be used to build safety and reliability into the design of high-temperature production of hydrogen (as opposed to back-fitting safety into current-generation nuclear power plants, which has proved to be costly).

The high-temperature hydrogen facility is really two plants in one, a nuclear power plant and a chemical processing plant. Balancing the needs of these two plants will require safety, cost, reliability, chemistry, and thermal-hydraulics models. The focus of the safety analysis work will be on reliability and safety, both of which will ultimately impact other attributes, such as cost and thermal-hydraulics. Consequently, this work (and the associated models) may be used to iterate on design characteristics specific to the facility in question.

The primary focus of the safety analysis work in this report is on determinating applicable separation distances to give the nuclear facility a sufficient buffer zone from the hazards associated with the hydrogen production plant, such as hydrogen gas detonations and toxic chemicals. Consequently, the distance evaluation is expressed as a distribution reflecting the probabilistic aspect of the analysis. Using this distance distribution, decision makers will be able to select a suitable level of protection to prevent unlikely events from impacting the nuclear facility. The analysis provides state-of-the-practice models, quantification, and recommendations specific to several issues, including:

- Separation distances from nuclear/hydrogen deflagration and detonation hazards
- Pressure wave calculations
- Blast loading calculations and failure probability quantifications
- Passive mitigation alternatives.

Separation distances from other process hazards, such as acid or other gas clouds dispersion, brings safety and reliability into the design picture for high-temperature production of hydrogen. Probabilistic methods (i.e., PSA) provides insights into the synergy between nuclear and chemical processes, answering safety-related questions specific to separations of the two processes.

1.2 Other Related Regulation and Past Analysis

Combustible gasses such as hydrogen, propane, acetylene, and methane are used in various capacities during normal operation of nuclear power plants. Hydrogen is one of the most prevalent combustible gasses and is safely being used at most nuclear power generation facilities. For example, hydrogen is used as a coolant for the main electrical generator in both pressurized water reactors (PWRs) and boiling water reactors (BWRs). Also, in PWRs, it is supplied to the chemical volume and control system to manage the dissolved oxygen concentration in the reactor coolant system. In BWRs, water chemistry control systems are used for the same purpose. And, for all light-water nuclear plants, hydrogen is present in the waste gas systems as a product of water dissociation during power operation.

Past hydrogen safety analyses focused on evaluating current generation plants with respect to:

- Location of potential leaks or upsets
- Rate of occurrence of deflagration and detonations
- Consequence of the rate, which is a function of inventory, venting, combustibility of gas, etc.

These analyses have found relatively low risk at current plants, due to such features as the distant separation of critical components from hazards (e.g., the hydrogen storage facility may be hundreds of

meters away from critical buildings), low inventory of gasses, abundance of backup safety systems, and passive safety systems such as flow-limiting valves. Since the high-temperature production of hydrogen plant will, by design, be relatively close to the nuclear facility and may generate and store a large quantity of hydrogen (and potentially other hazardous chemicals), it is critical that safety be evaluated and controlled early in the design process for this facility.

Generic Safety Issue (GSI) 106, Piping and the Use of Highly Combustible Gasses in Vital Areas, was identified in NUREG-0705 by the NRC in 1981 as a potential issue warranting attention (Simion et al. 1993). This GSI addressed the issue of potential damage resulting from the use of combustible gasses such as hydrogen, propane, acetylene, and methane in operating nuclear power plants. While the distribution and storage systems for these gasses are not, in many cases, safety related, it was apparent that leakage or ruptures from failures of these systems in areas containing or adjacent to safety-related equipment have the potential to impact overall plant risk. Originally, the GSI-106 technical work focused only on hydrogen piping inside safety-related areas. The scope was later expanded to include such concerns as hydrogen storage facilities.

Current NRC regulations and guidelines consider the use and storage of highly flammable gasses (such as hydrogen) and liquids within and near the plant. Specifically, 10 CFR 50, Appendix R, regulations explicitly require plants to establish administrative controls that "prohibit the storage of combustibles in safety-related areas or establish designated storage areas with appropriate fire protection." Standard Review Plan (SRP), Section 9.5.1, iterates these control measures and recommends design and safety features of hydrogen systems. More recent “risk-informed” types of NRC regulations are stated in 10 CFR 50.44 and associated guidance documents such as Regulation Guide (RG) 1.174.

General guidance on gaseous hydrogen storage may be found in national fire protection standards such as in Gaseous Hydrogen Systems at Consumer Sites, NFPA 50A. For standards concerning *liquefied* hydrogen storage, “Liquefied Hydrogen Systems at Consumer Sites,” NFPA 50B, is applicable. However, due to the costs associated with operation, we do not expect that liquefied storage of hydrogen will be used in any HTGR-to-hydrogen production facilities.

Other pertinent rules include the OSHA Regulations (Standards, 29 CFR), which for hydrogen is in Part 1910, Section 103 (Hydrogen, 1910.103). Note that OSHA considers hydrogen a hazardous material and governs if accordingly. Any industrial facility that stores and distributes hydrogen must conform to OSHA Part 1910.103(b)(1)(iii). This standard provides guidance on such construction details as joints in piping and tubing (e.g., they may be made by welding or brazing or by use of flanged, threaded, socket, or compression fittings). Further, crude guidance is provided on separation distances in 1910.103(b)(2)(ii). This section states that the location of a hydrogen system must be a minimum of 50 ft from intakes of ventilation or air-conditioning equipment and air compressors. From a risk perspective, however, hydrogen storage facilities may need to be much farther away from risk-sensitive buildings and components.

Storage containers for hydrogen or other combustible gasses must comply with the following:

- ASME Boiler and Pressure Vessel Code, Section VIII - Unfired Pressure Vessels
- API Standard 620, Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks
- OSHA Appendix R Section 1910.6.

Any facility using hydrogen gas must comply with the National Fire Protection Association (NFPA) Codes and Standards, NFPA 50A and NFPA 220. NFPA 50A is the standard for gaseous hydrogen systems at consumer sites; NFPA 220 is the standard on types of building construction. These standards

dictate the methods required for construction and use of hydrogen. However, these standards provide only minimal requirements—agencies such as the NRC may require more strict requirements, as needed.

Current NRC standards for combustible gas activities fall under the technical requirements of 10 CFR Part 50, specifically 10 CFR 50.44 (Standards for Combustible Gas Control System in Light Water-Cooled Power Reactors). In conjunction with these deterministic standards, a risk-informed framework has also been recently adopted and used in NRC activities. The NRC staff has developed a framework that describes the overall approach, process, and guidelines they will use when reviewing risk-informed alternatives to the deterministic requirements stated in 10 CFR Part 50. Initially, this framework was part of SECY-00-0086. However, the risk-informed applications have expanded to numerous other parts of the NRC regulations (e.g., CFR 50.46, RG 1.174, CFR 50.109). Guiding principles to this modern approach to regulation at the NRC include:

1. Use a risk-informed defense-in-depth approach to accomplish the goal of protecting public health and safety.
2. Set guidelines that limit
    a. The frequency of accident initiating events
    b. The probability of core damage
    c. The release magnitude following core damage events
    d. The effects to the public during accidents
3. Consider both design-basis and core-melt accidents
4. Treat applicable uncertainties
5. Use quantitative guidelines based on the NRC Safety Goals and the subsidiary objectives of 1×10^{-4} per reactor year for core damage frequency and 1×10^{-5}/ry for large early-release frequency.

1.3 Analysis Safety Criteria

Current NRC risk-informed applications use the criteria in RG 1.174. An overview of this process is depicted in Figure 3. Following this process, the general principles of risk-informed regulation followed by the NRC include:

- The application meets current regulations unless it explicitly relates to a requested exemption or rule change
- The application is consistent with the general defense-in-depth philosophy
- The application maintains sufficient safety margins
- The application maintains small risk and is consistent with the intent of the NRC's Safety Goal Policy Statement.

Of importance for next-generation nuclear facilities is the second bullet above, specifically that relating to the application of defense-in-depth. Even though advanced facilities may be demonstrated to be safer than the current generation of plants, the NRC may still require some level of defense-in-depth to account for the uncertainty inherent in the safety of plant operations.

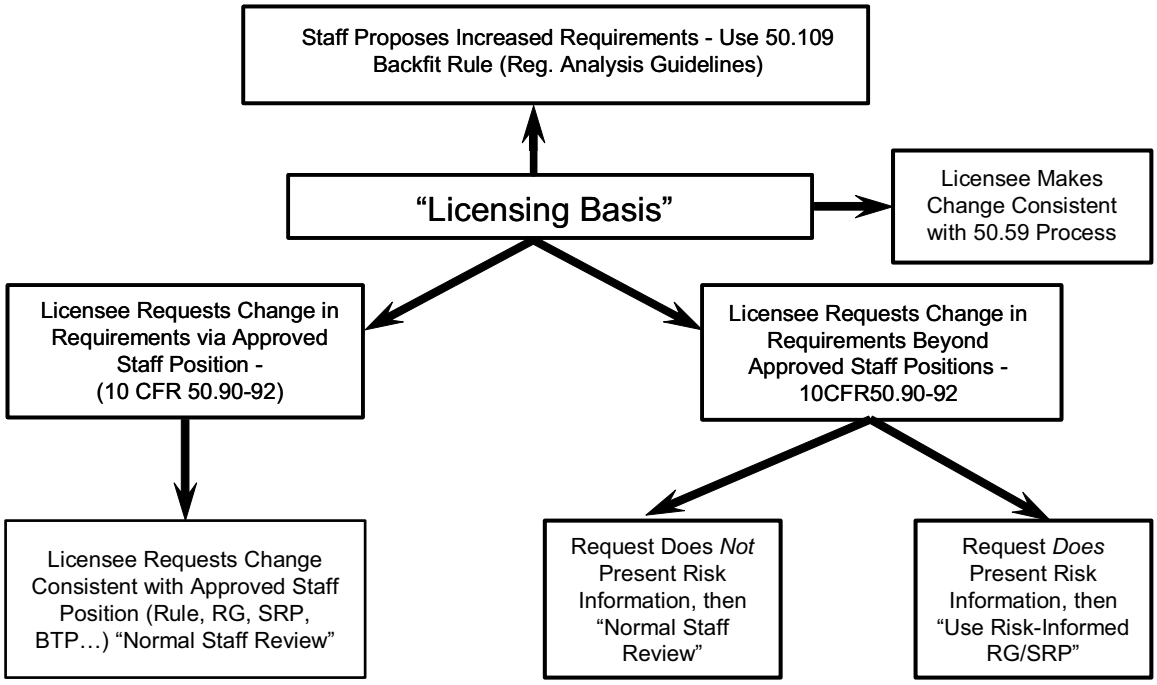

Figure 3. Typical path for regulating nuclear power plants by the NRC, including risk-informed applications.

When issues are evaluated using RG 1.174 at the NRC, all safety impacts of the issue are to be evaluated in an integrated manner in order to improve operational and engineering decisions. Quantifications of risk is a fundamental part of this process. Toward that goal, the NRC has indicated that metrics such as core damage frequency and large early release frequency may be used as bases for PSA. However, they have specifically requested that appropriate consideration of uncertainty be given in the analyses and that an interpretation of findings be performed as part of any analysis.

Decision criteria have been identified as part of RG 1.174 in order to allow or disallow changes to existing nuclear power plants. These criteria are shown in Figure 4. While the figure has only been applied for the current generation of nuclear power plants, we would expect that risk from next-generation plants would (a) be held to a stricter limit and (b) would be evaluated in similarly. What can be taken from the information in Figure 4 is that a core damage frequency (for the proposed hydrogen facility) that is above the 1E-6/yr threshold would be scrutinized. In a follow-on to this preliminary report, we will evaluate a variety of risks in terms of potential core damage—these risk estimates will be compared to the decision criteria spelled out in RG 1.174 and shown in Figure 4.

Other applicable regulations that potentially play a role in the design, construction, and operation of the hydrogen production facility are RG 1.91, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (NRC 1978) and RG 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (NRC 2001a).

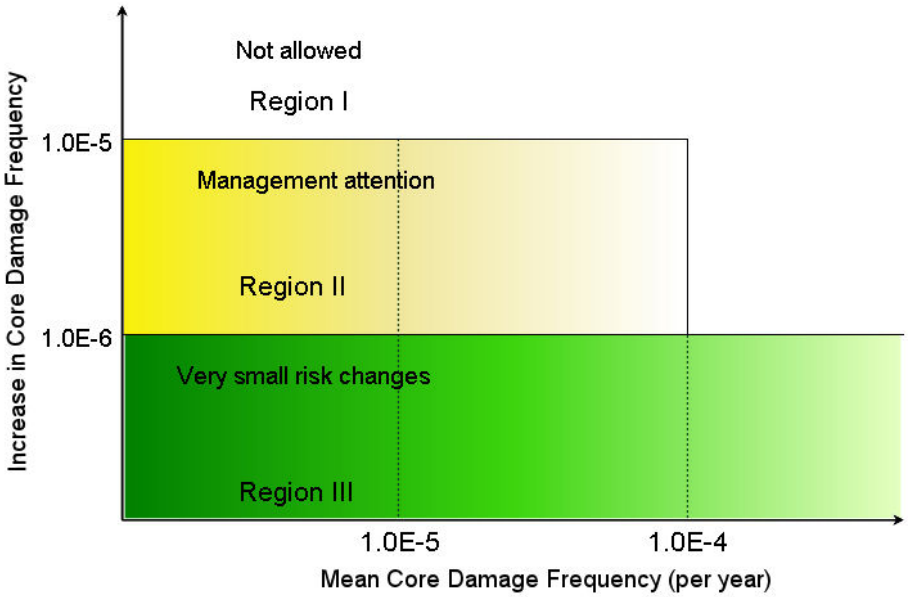

Figure 4. Decision criteria for risk-informed applications at the NRC (from RG 1.174).

Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," was released in 1978. This guide suggests that overpressures from explosions should not exceed 7 kPa on plant structures. Based on experimental TNT data, a safe distance can then be defined by the relationship $R > 18\ W^{1/3}$, where R is the distance (in meters) from an exploding charge of W pounds of TNT (or equivalent gas such as hydrogen). However, we did not use approximate methods such as TNT equivalencies in our analysis. Instead, we used the results from a computer code that employs physics-based principles to model the shock waves and combustion processes inside a reacting hydrogen mass.

Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, was updated and released in 2001. This guide discusses issues relating to the protection of nuclear power plant control rooms. It suggests that the control room should be adequately protected from chemical dispersions outside the plant. While primarily focusing on chemical sources such as storage tanks, railroad cars, barges, etc., the utilization of hazardous chemicals on site as part of hydrogen production falls under the umbrella of this regulatory guide. The guide further specifies criteria for control room habitability:

> "Any hazardous chemical stored onsite within 0.3 miles [482 m] of the control room in a quantity greater than 100 pounds [45 kg] should be considered for control room habitability evaluation. Hazardous chemicals should not be stored within the close proximity (generally within 330 feet [91 m] or less) of a control room or its fresh air inlets, including ventilation system intakes and locations of possible infiltration such as penetrations. Small quantities for laboratory use, 20 pounds [9 kg] or less, are exempt. The maximum allowable inventory in a single container stored at specified distances beyond 330 feet [91 m] from the control room or its fresh air inlet varies according to the distance and the control room type" (NRC 2001a).

Our dispersion analysis (Appendix A) provides a detailed evaluation of the hazardous limits of a variety of chemicals that could be released under accident scenarios. We did not, however, perform

control room habitability studies relating to the potential intake of chemicals into the control room, filtering mechanisms, and air flows through postulated control room structures. In order to perform this type of detailed analysis, the design of both the nuclear and chemical facilities would need to be better specified, for example the location and type of the control room would need to be known (e.g., above/below ground, single control room for multiple reactors, enclosure/filtering type), and the quantities and composition of hazardous chemicals would need to be listed.

2. PROBABILISTIC ANALYSIS

2.1 Introduction

Since the facilities discussed in this report (both the nuclear and chemical portions) are still in the preliminary stages, a variety of design assumptions were necessary for the evaluation. Supplementing these assumptions, sensitivity calculations were performed (e.g., varying the separation distance between the nuclear and chemical plants; varying the locations of key systems, such as the control room). While it is expected that extremely large quantities of hydrogen will *not* be stored permanently on site, an appreciable amount of high-pressure gaseous hydrogen is expected to be near the nuclear facility. Further, different chemical processes will entail different types and quantities of potentially hazardous material. Thus, calculations were performed on the chemical process to be used, including:

- A sulfur-iodine process with sulfuric acid, I2, and HI
- A hybrid sulfur process with sulfuric acid but no iodine compounds
- A high-temperature electrolysis process without hazardous chemical feedstocks.

Our PSA work focused on a few key areas to determine adequate separation between the nuclear and chemical facilities. Figure 5 outlines the areas identified for consideration in this report. In general, two cases, overpressure events (e.g., detonations and deflagrations) and dispersion events describe the potential accident scenarios. Note that we did not perform a PSA on the HTGR itself; that is, events internal to the HTGR are excluded from this analysis. Nor did we analyze risk implications due to the secondary heat exchanger. For example, the choice of the secondary working fluid, molten salt or helium, was excluded because these will not cause external impacts to the HTGR, with the exception of potentially causing a reactor transient, which falls into the category of HTGR internal events. However, interactions between components in the chemical facility that may increase risk to the HTGR (e.g., causing a hydrogen explosion) are included in the scope of the analysis.

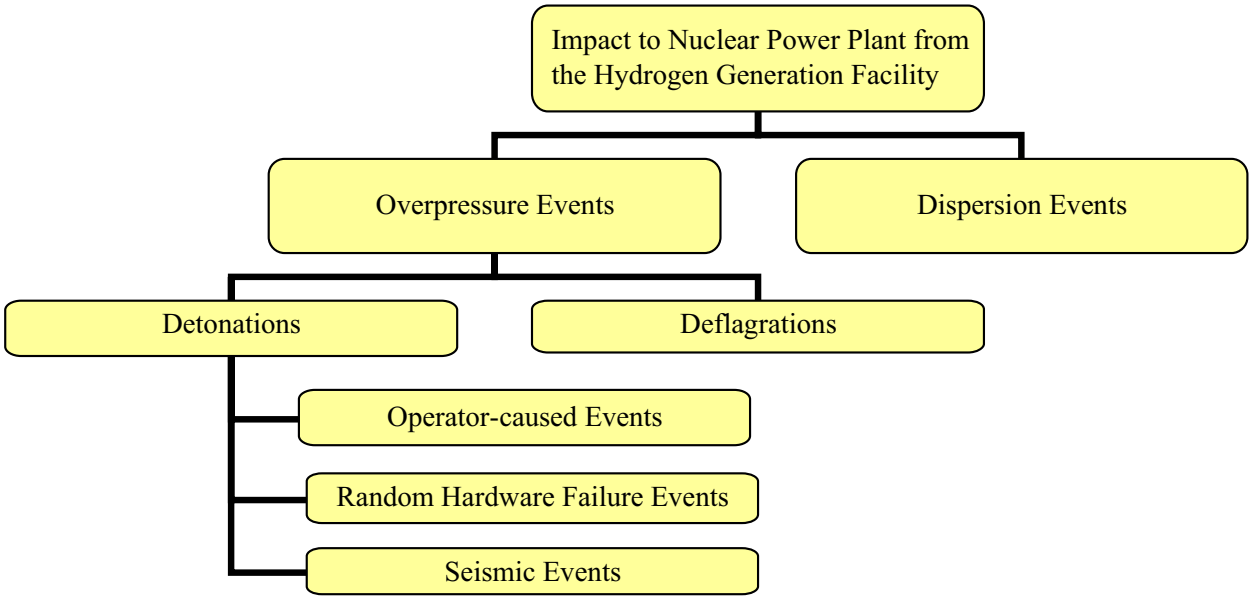

Figure 5. Master logic diagram for potential disruption scenarios.

As mentioned, we identified and analyzed dispersion cases for a set of representative hazardous chemicals. For these dispersion models, we selected the ALOHA code developed by the EPA and NOAA. The analysis for dispersion events is presented in Appendix A. The frequency of these potential releases is evaluated in this section. To evaluate the risk from hydrogen detonation and deflagration events, we selected the SAPHIRE risk analysis code developed at INEEL (Russell et al. 1999). The SAPHIRE code was supplemented with associated modeling such as hydrogen blast wave calculations.

To model the detonation events, we constructed scenarios specific to each potential accident category. These models use the event tree and fault tree capabilities provided by the SAPHIRE software. Since the facility being analyzed has not yet been fully designed, we assumed likely configurations, layouts, and operational practices to perform the analysis.

2.2 Modeling the Accident Scenarios

The scenario modeling was performed using the probabilistic risk assessment code SAPHIRE (The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations) which refers to a set of microcomputer programs developed to create and analyze PSAs.

2.2.1 Event Tree Modeling

The facility scenario modeling involves the representation of three initiating events:

1. Random hydrogen explosions
2. Human-induced hydrogen upsets
3. Seismic-induced acid or hydrogen rupture.

From these initiating events, plant response to these initiating events are represented using event trees with various end states of the facility response to the initiator.

2.2.1.1 Random Hydrogen Explosion Event Tree

The random hydrogen explosion event tree represents the nuclear power plant response to a random hydrogen explosion event. Figure 6 shows the random hydrogen explosion event tree.

The initiating event for this event tree is called DETONATION. This event represents a random explosion in the hydrogen production plant. The two top events for the event tree are STRUCTURAL and CORE_DAMAGE. The STRUCTURAL top event represents a single basic event system that models the probability of the nuclear power plant sustaining damage given there is a hydrogen explosion. This event is further developed in Section 2.3.1. The CORE_DAMAGE top event represents a single basic event that models core damage to the nuclear power plant given that there was structural damage to the nuclear plant. This event is further developed in Section 2.3.3. The end state outcomes for the random hydrogen explosion event tree are DISPERSION, CD, or DAMAGE. The DISPERSION end state represent those scenarios that result in a dispersion of chemicals. The core damage end state or CD represents those scenarios that result in damage to the nuclear fuel in the core. The DAMAGE end state represents those scenarios that result in nuclear plant structure being impacted but do not result in damage to the core.

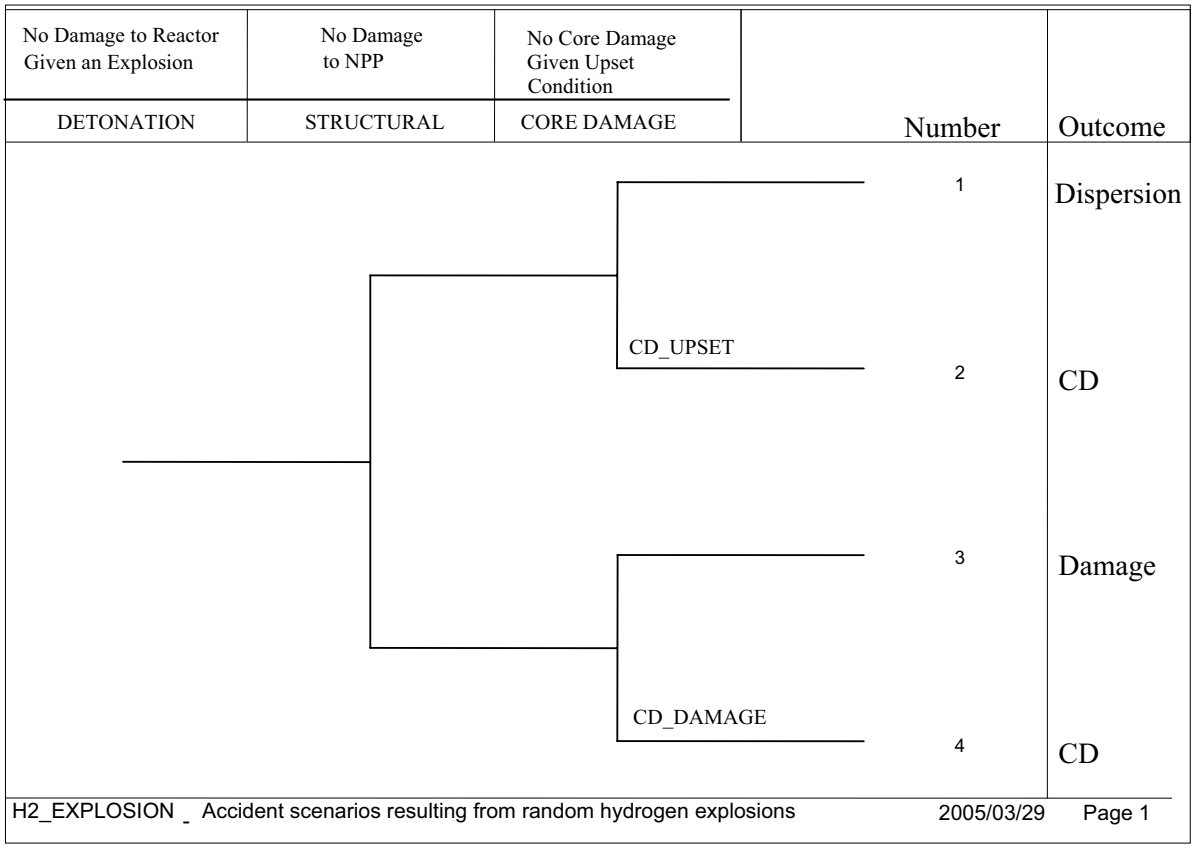

Figure 6. Random hydrogen explosion event tree.

The event tree structure is used to represent the interaction among the top events when exposed to the initiating event. If there is a hydrogen explosion, then the structural damage of the power plant is questioned. If there is no structural damage with STRUCTURAL being successful (the up branch), there may still be a possibility of core damage given that an upset condition did occur. If CORE_DAMAGE is successful (the up branch), there could still be present an hazardous chemical leak from the chemical plant, which gives Sequence 1, having an end state of DISPERSION. If CORE_DAMAGE has occurred given an upset condition, then this scenario leads to Sequence 2, having an end state of CD.

If there is a hydrogen explosion that results in structural damage with STRUCTURAL being failed (the down branch), then again the core damage is questioned. If there is no core damage, with CORE_DAMAGE being successful (the up branch), then this results in Sequence 3, having an end state of DAMAGE. With this end state, there is still damage to nuclear plant and a chemical leak, but there is no core damage to the nuclear fuel. If there is structural damage and there is core damage with CORE_DAMAGE being failed (the down branch), then this leads Sequence 4, having an end state of CD.

2.2.1.2 Human-Caused Hydrogen Explosion Event Tree

The human-caused hydrogen explosion event tree represents the nuclear power plant response from a human instigated hydrogen explosion event. Figure 7 shows the human-caused hydrogen explosion event tree.

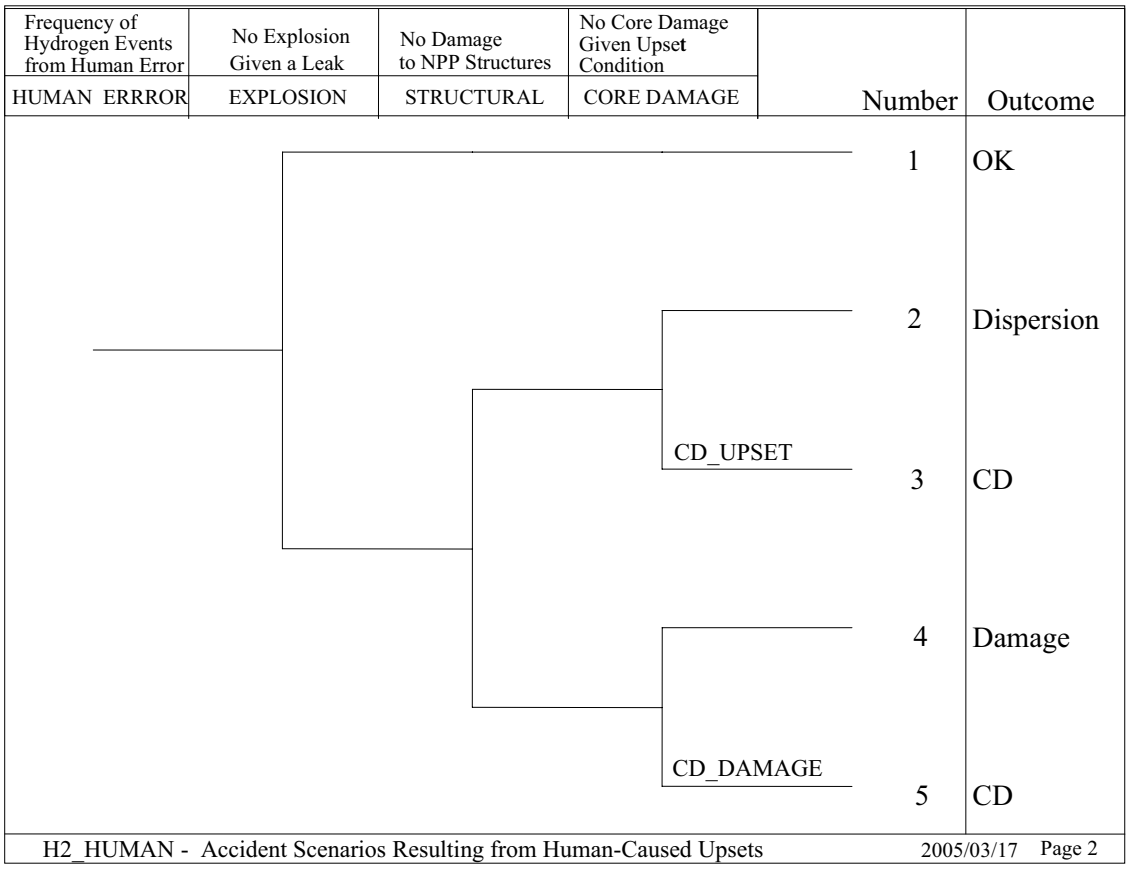

Figure 7. Human-caused-upsets event tree.

The initiating event for this event tree is HUMAN ERROR. This event represents a human error-caused explosion in the hydrogen storage plant. The three top events for the event tree are EXPLOSION, STRUCTURAL, and CORE_DAMAGE. The EXPLOSION top event is a single basic event system that represents the probability of a hydrogen explosion given a hydrogen leak. The STRUCTURAL top event represents a single basic event system that models the probability of the nuclear power plant sustaining damage given there is a hydrogen explosion. The CORE_DAMAGE top event represents a single basic event that models core damage to the nuclear power plant given that there was structural damage to the nuclear plant. The end state outcomes for the human-caused hydrogen explosion event tree are DISPERSION, CD, or DAMAGE. The DISPERSION end state represents those scenarios that result in a dispersion of chemicals. The core damage end state or CD represents those scenarios that result in damage to the nuclear fuel in the core. The DAMAGE end state represents those scenarios that result in the nuclear plant structure being impacted but do not result in damage to the reactor core.

The event tree structure is used to represent the interaction among the top events when exposed to the initiating event. If a human error-induced hydrogen leak event occurs, then the possibility of an explosion occurring from the leak is questioned. If there in no explosion with the EXPLOSION top event being successful (the up branch), then this results in Sequence 1, having an end state of OK where no substantial negative consequences occur. If there is a hydrogen explosion, then the structural damage of the power plant is questioned. If there is no structural damage with STRUCTURAL being successful, there may still be a possibility of core damage given that an upset condition did occur. If CORE_DAMAGE is successful (the up branch), there could still be present an acid leak from the

chemical plant, which gives Sequence 2, having an end state of DISPERSION. If CORE_DAMAGE has occurred given an upset condition, then this scenario leads to Sequence 3, having an end state of CD.

If there is a hydrogen explosion that results in structural damage with STRUCTURAL being failed (the down branch), then again the core damage is questioned. If there is no core damage, with CORE_DAMAGE being successful (the up branch), then this gives Sequence 4, having an end state of DAMAGE. With this end state, there is still damage to nuclear plant and an acid leak, but there is no core damage to the nuclear fuel. If there is structural damage and there is core damage with CORE_DAMAGE being failed, then this leads to Sequence 5, having an end state of CD.

2.2.1.3 Seismic-Induced Hydrogen Explosion Event Trees

The seismic-induced hydrogen explosion event trees represent the nuclear power plant response from a seismic-induced hydrogen explosion event. The seismic-induced hydrogen explosion is represented by three event trees, with each event tree signifying a seismic initiating event at a specific g-level ground acceleration (i.e., earthquake magnitude). For each g-level earthquake initiator, the top event EQ-OCCUR is questioned if an earthquake at that particular g-level occurred. If the earthquake did not occur, resulting end state 1 is OK with no plant damage. If the earthquake at the specified g-level did occur and EQ-OCCUR is failed, then this results in a transfer to the event tree SEIS1.

Figure 8 shows the seismic g-level 1 hydrogen explosion event tree. For this event tree, the g-level ground acceleration (i.e. earthquake magnitude) is 0.1. At this g-level, the ground acceleration is low and not many components are expected to fail due to seismic ground motion.

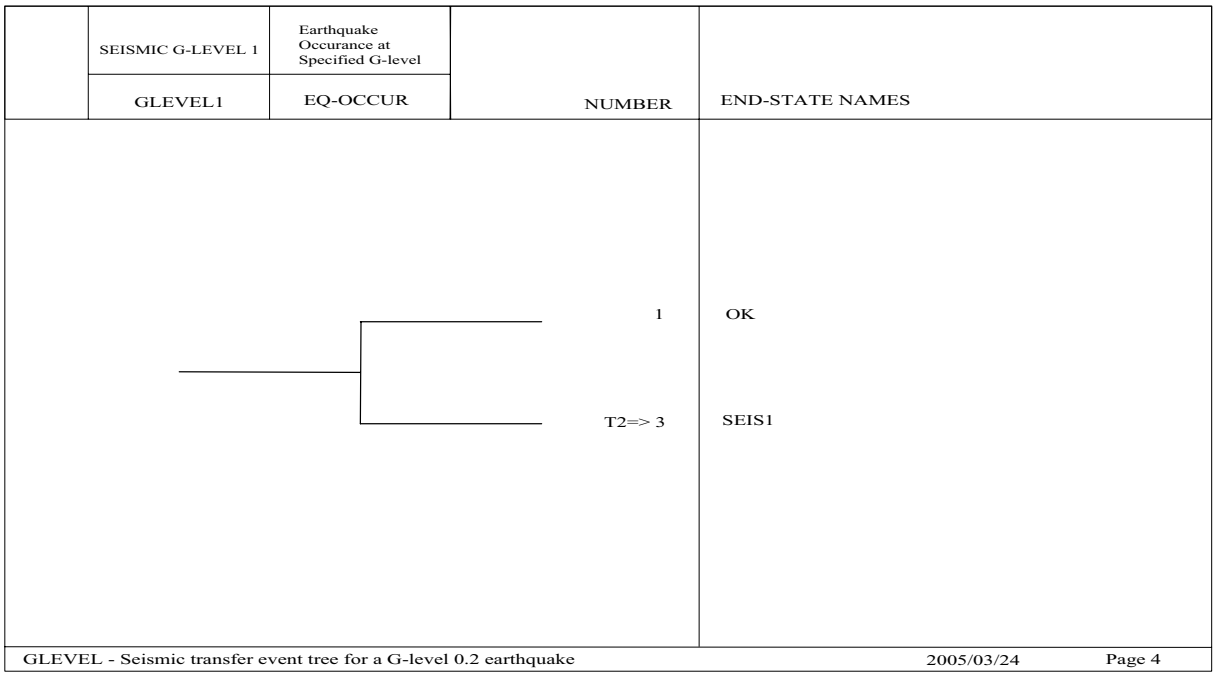

Figure 8. G-Level 1 seismic event tree.

Figure 9 shows the seismic g-level 2 hydrogen explosion event tree. For this event tree, the g-level ground acceleration (i.e., earthquake magnitude) is 0.3. At this g-level, ground acceleration is significant and has the likelihood of several components failing due to seismic ground motion increases.

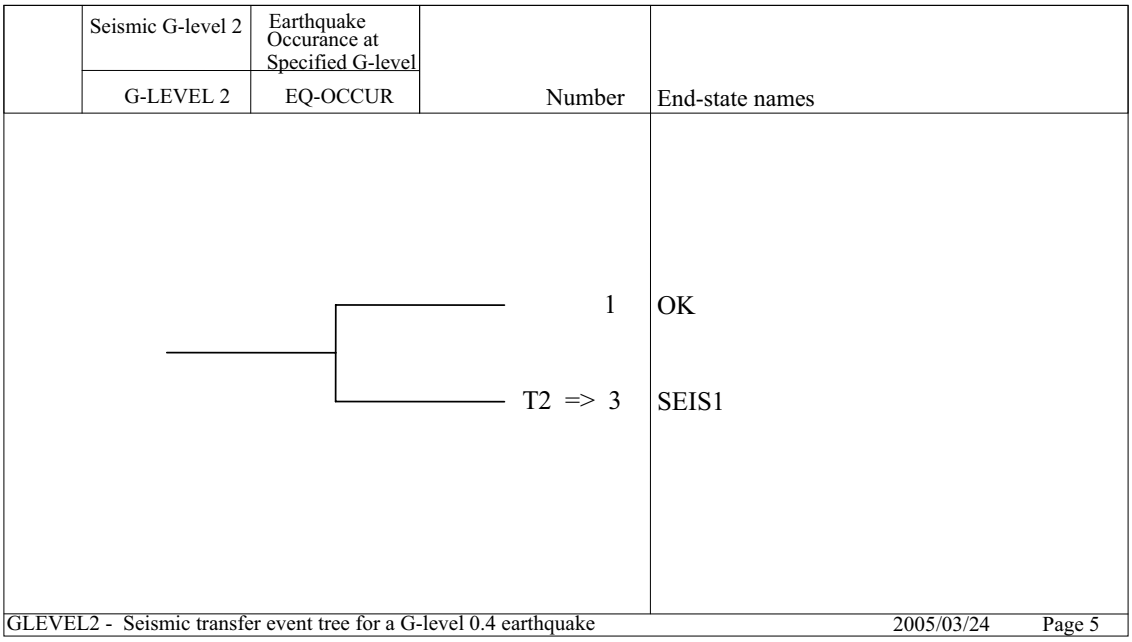

Figure 9. G-Level 2 seismic event tree.

Figure 10 shows the seismic g-level 3 hydrogen explosion event tree. For this event tree, the g-level ground acceleration (i.e., earthquake magnitude) is 0.6. At this g-level, ground acceleration is high, and the likelihood of several components failing due to seismic ground motion is the greatest.

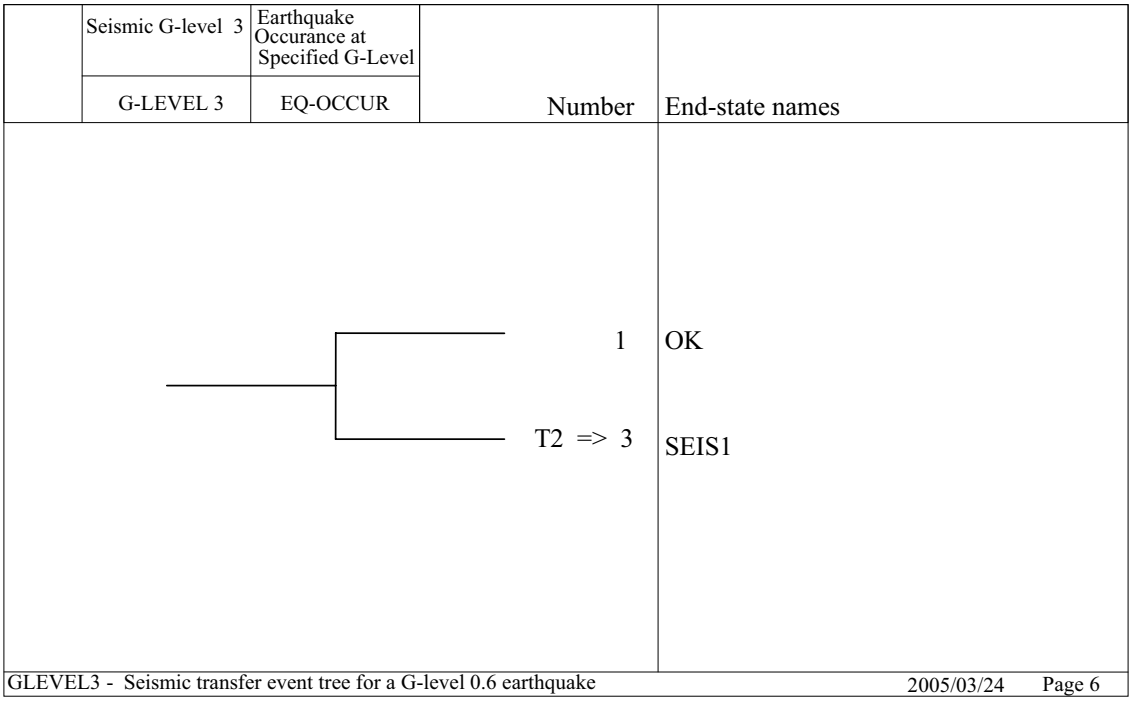

Figure 10. G-Level 3 seismic event tree.

Once the g-level of the earthquake has been determined, each seismic g-level event tree transfers to the main seismic-induced hydrogen explosion event tree, SEIS1. The SEIS1 event tree represents the nuclear power plant response from a seismic-induced hydrogen explosion event. Figure 11 shows the main seismic-induced hydrogen explosion event tree.

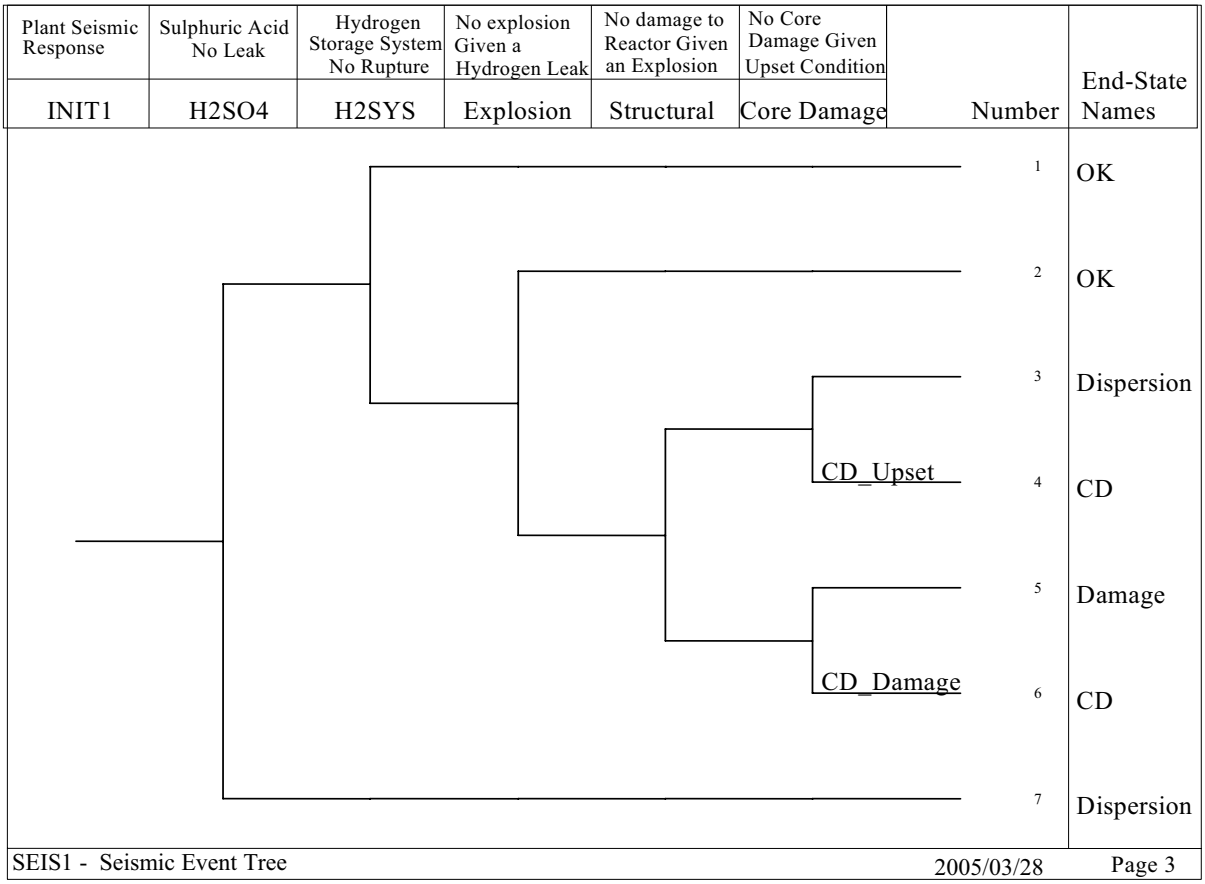

Figure 11. Seismic-induced hydrogen explosion event tree.

The initiating event for this event tree is determined by the G-level event trees and is either GLEVEL1, GLEVEL2, or GLEVEL3. These initiating events represent the probability of occurrence of a specified ground acceleration earthquake (g-level). If a specified g-level earthquake occurs, the resulting end state is a transfer to the SEIS1 event tree. The first top event in this tree is INIT1, which is an event that represents the occurrence of specific g-level earthquake. The remaining top events for the event tree are H2SO4, H2SYS, EXPLOSION, STRUCTURAL, and CORE_DAMAGE. The H2SO4 top event represents the success or failure of the sulfuric acid plant system. Success of this system is the non-occurrence of a hazardous chemical leak. The H2SYS top event represents the success or failure of the hydrogen production system. Success of this system is nonoccurrence of the discharge of hydrogen due to a rupture. The EXPLOSION top event is a single basic event system that represents the probability of a hydrogen explosion given a hydrogen leak. The STRUCTURAL top event represents a single basic event system that models the probability of the nuclear power plant sustaining damage given there is a hydrogen explosion. The CORE_DAMAGE top event represents a single basic event that models core damage to the nuclear power plant given that there was structural damage to the nuclear plant.

The end state outcomes for the seismic-induced hydrogen explosion event tree (SEIS1) are DISPERSION, CD, or DAMAGE. The DISPERSION end state represents those scenarios that resulted in a dispersion of chemicals. The core damage end state or CD represents those scenarios that resulted in damage to the nuclear fuel in the core. The DAMAGE end state represents those scenarios that resulted in the nuclear plant structure being impacted but did not result in damage to the core.

The event tree structure is used to represent the interaction among the top events when exposed to the seismic initiating event. If specific g-level earthquake occurs, then the condition of sulfuric acid plant is questioned. If the earthquake caused an acid leak with H2SO4 being failed, then this scenario results in Sequence 7, having an end state of DISPERSION. If the sulfuric acid plant remains intact with no acid leak, where H2SO4 is successful, then the condition of the hydrogen storage system is questioned next. If the hydrogen storage system remains intact, where H2SYS is successful, then the result is Sequence 1, having an end state of OK, where no substantial negative consequences occur. If the hydrogen storage system is damaged by the seismic event with H2SYS being failed, this results in the top event EXPLOSION (the possibility of an explosion occurring from a hydrogen leak) being questioned.

If there in no explosion with EXPLOSION being successful, then this scenario results in Sequence 2, having an end state of OK, where no substantial negative consequences occur. If there is a hydrogen explosion, then the structural damage of the power plant is questioned. If there is no structural damage with STRUCTURAL being successful, there may still be a possibility of core damage given that an upset condition did occur. If CORE_DAMAGE is successful, there could still be present an acid leak from the chemical plant, which gives Sequence 3, having an end state of DISPERSION. If CORE_DAMAGE has occurred given an upset condition, then this scenario leads to Sequence 4, having an end state CD.

If there is a hydrogen explosion that results in structural damage with STRUCTURAL being failed, then again the core damage is questioned. If there is no core damage, with CORE_DAMAGE being successful, then this gives Sequence 5, having an end state of DAMAGE. With this end state, there is still damage to nuclear plant and an acid leak, but there is no core damage to the nuclear fuel. If there is Structural damage and there is core damage with CORE_DAMAGE being failed, then this leads Sequence 6, having an end state of CD.

2.2.2 System Fault Tree Modeling

This section presents the fault tree models required to support the top events in the event trees. In many cases, the system top events are represented by a single basic event. The sulfuric acid system and hydrogen storage system are systems with multiple components, and these systems require representation with more complicated fault trees.

The following assumptions and guidelines were used in developing the system fault tree models:

- Only the failure of major components that could rupture or leak (i.e., pumps, valves, cooling units) was considered
- The component failure mode for all components in the hydrogen storage system and sulfuric acid system is failure due to rupture or leak
- Possible human recovery actions from the leak or rupture were not modeled
- Rupture or leak due to common-cause failures were not considered.

2.2.2.1 Hydrogen Storage System Fault Tree Model

The hydrogen storage system is a highly pressurized system designed to pressurize hydrogen gas using compressors and store the gas into pressure vessels. Accumulators are used to take the pressurized hydrogen from pressure vessels and disperse it for use or transportation. Figure 12 illustrates a simplified, standard hydrogen storage system. This system is what is used to develop the hydrogen storage system fault tree model.

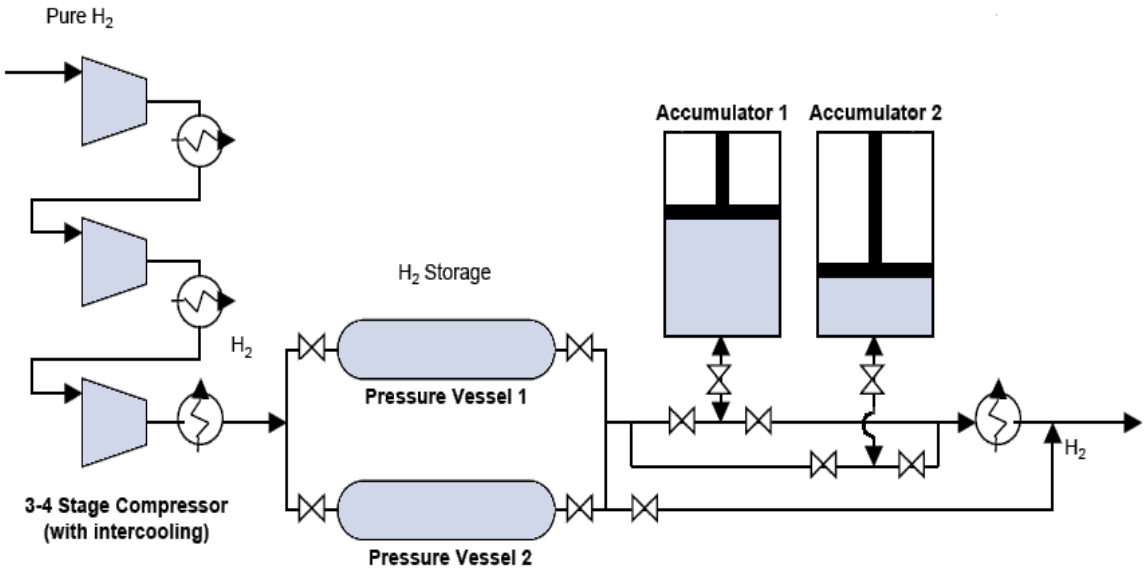

Figure 12. Hydrogen storage system piping diagram.

The hydrogen system is represented by the H2SYS-S1, H2SYS-S2, and H2SYS-S3 fault trees. These fault trees contain the random rupture failure events for the major components and the seismic-induced failures for the seismic susceptible components. Figure 13 illustrates fault tree H2SYS-S1. This fault tree contains seismic events for a 0.1 g-level earthquake and is used in the GLEVEL1 seismic event tree. Figure 14 illustrates fault tree H2SYS-S2. This fault tree contains seismic events for a 0.3 g-level earthquake and is used in the GLEVEL2 seismic event tree. Figure 15 illustrates fault tree H2SYS-S3. This fault tree contains seismic events for a 0.6 g-level earthquake. For all three fault trees, the shaded components in the figures represent the seismic-susceptible events.

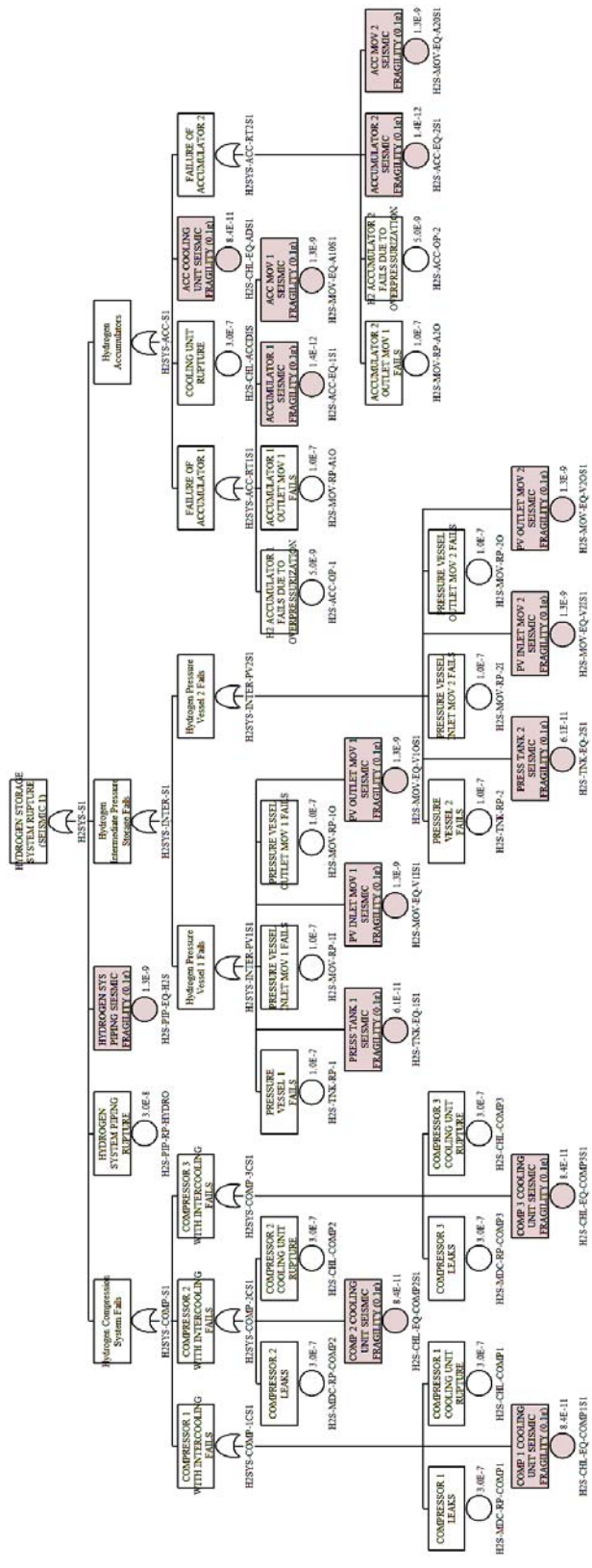

Figure 13. Fault tree H2SYS-S1.

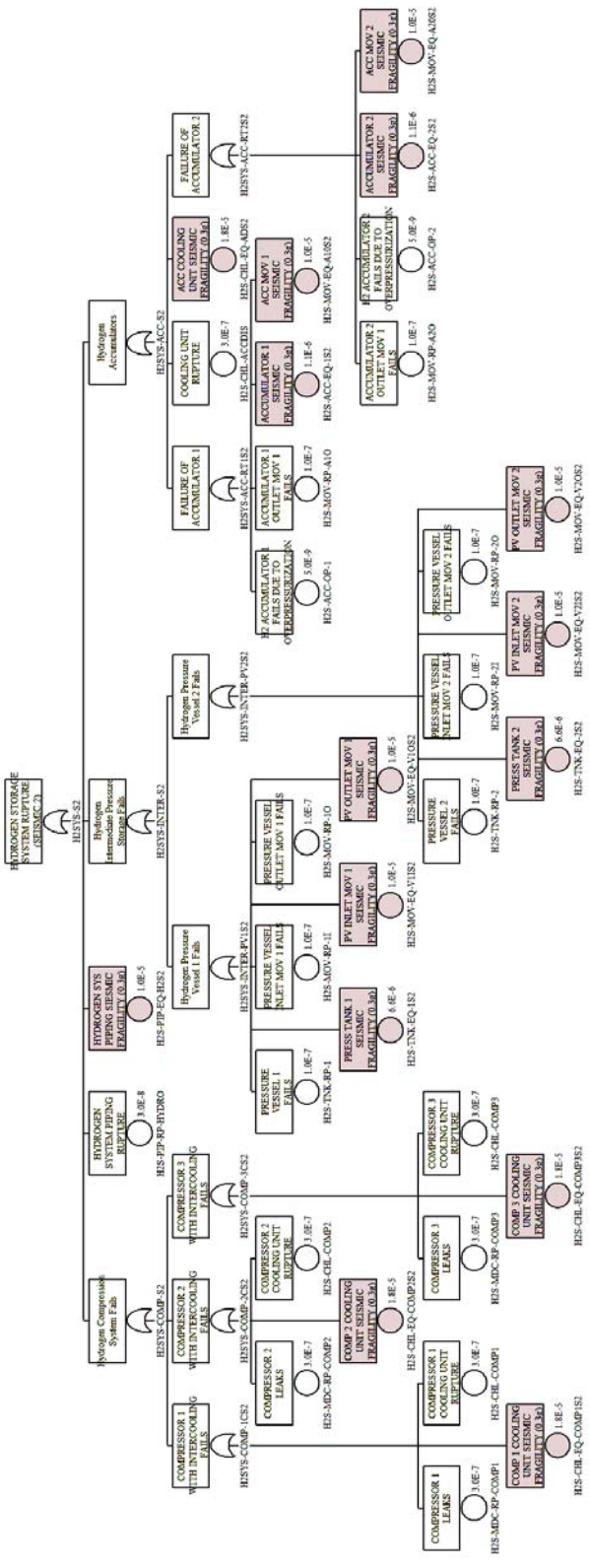

Figure 14. Fault tree H2SYS-S2.

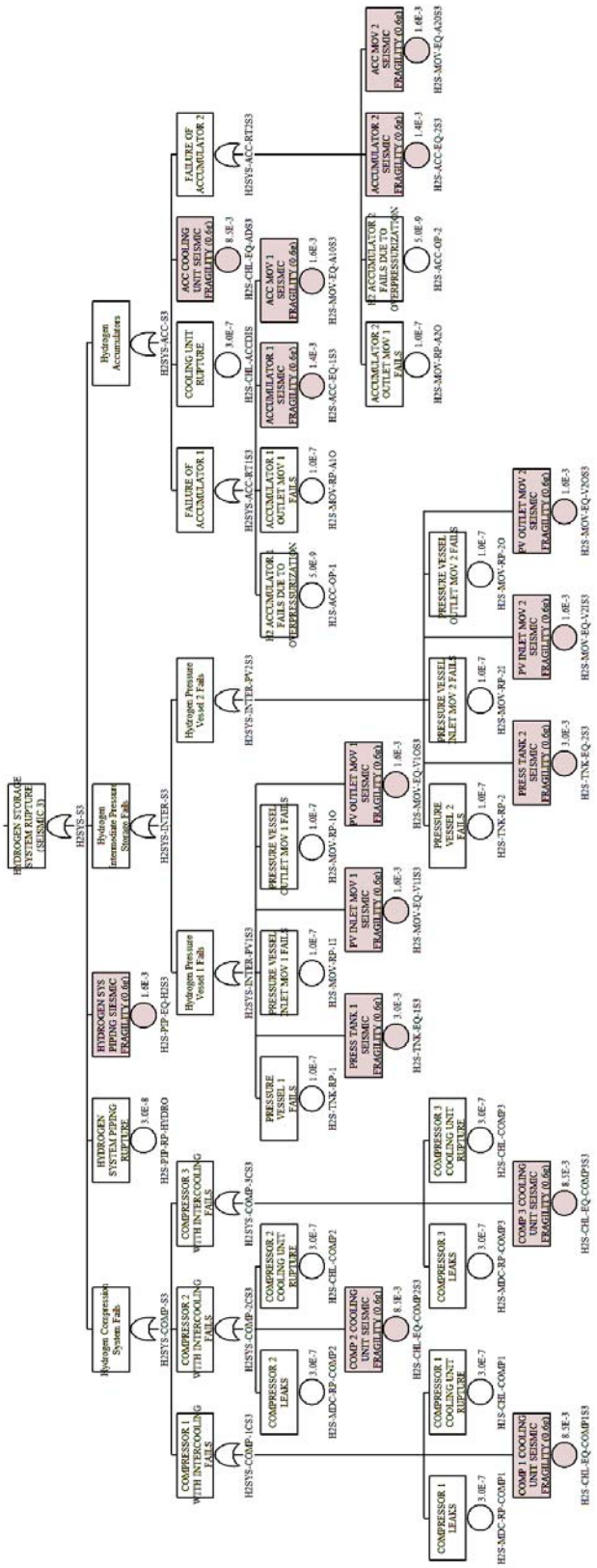

Figure 15. Fault tree H2SYS-S3.

2.2.2.2 Sulfuric Acid System Fault Tree Model

A sulfuric acid system is designed to take raw elemental sulfur and create sulfuric acid for storage in a large storage tank. Figure 16 illustrates a simplified, generic sulfuric acid plant. The major components modeled were the air blower, combustion chamber, acid pump, and final product storage tank. We used this system depiction as a surrogate for the chemical facility that will employ hazardous chemicals.

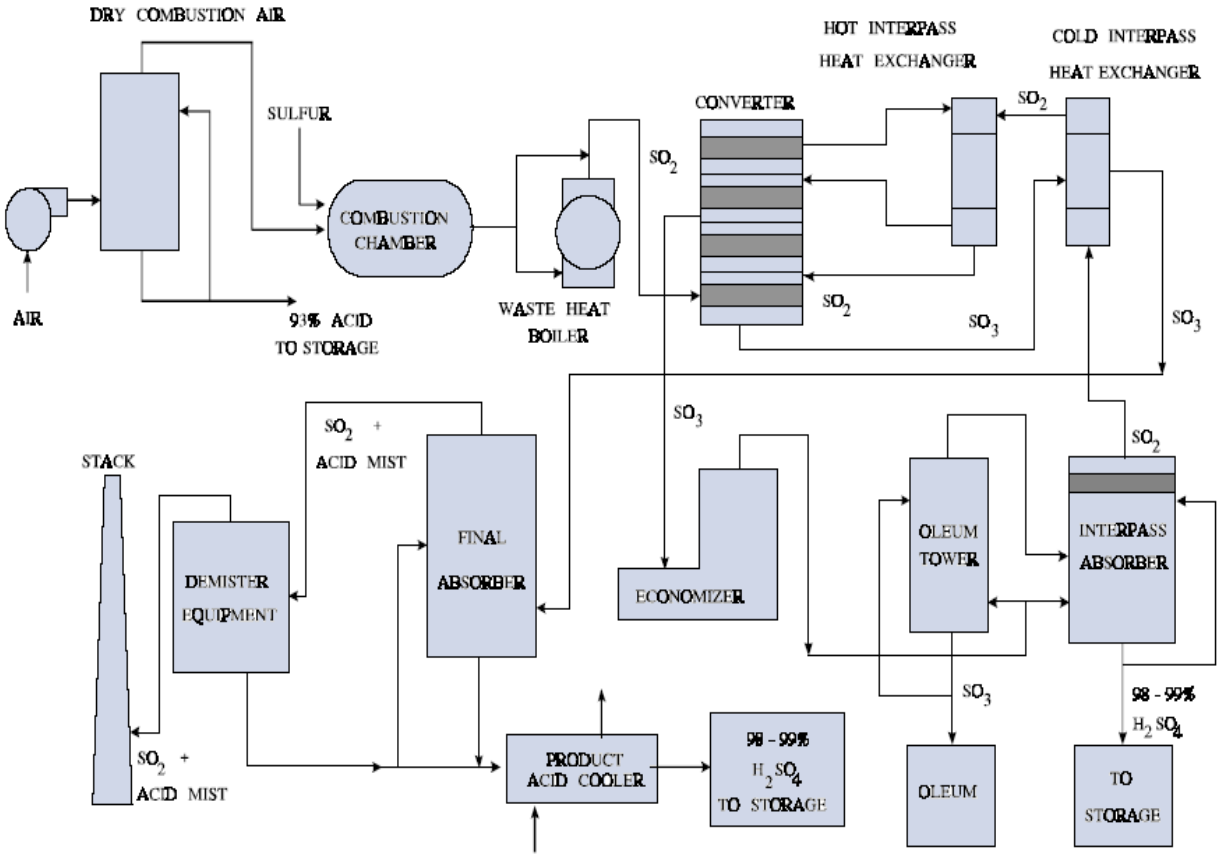

Figure 16. Sulfuric acid system piping diagram.

The sulfuric acid system is represented by the H2SO4-S1, H2SO4-S2, and H2SO4-S3 fault trees. These fault trees contain the random rupture failure events for the major components and the seismic-induced failures for the seismic susceptible components. Figure 17 illustrates fault tree H2SO4-S1. This fault tree contains seismic events for a 0.1 g-level earthquake and is used in the GLEVEL1 seismic event tree. Figure 18 illustrates fault tree H2SO4-S2. This fault tree contains seismic events for a 0.3 g-level earthquake and is used in the GLEVEL2 seismic event tree. Figure 19 illustrates fault tree H2SO4-S3. This fault tree contains seismic events for a 0.6 g-level earthquake. For all three fault trees, the shaded components in the figures represent the seismic susceptible events.

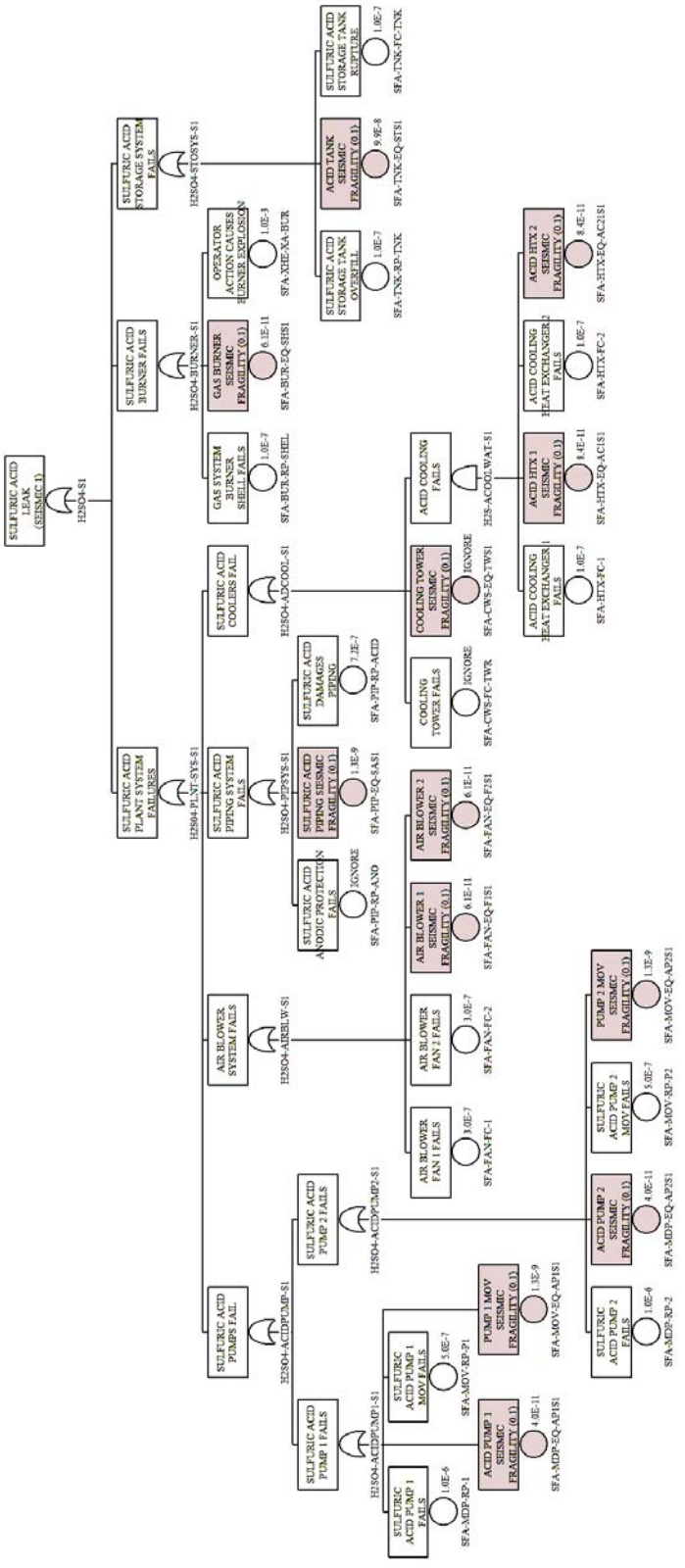

Figure 17. Fault tree H2SO4-S1.

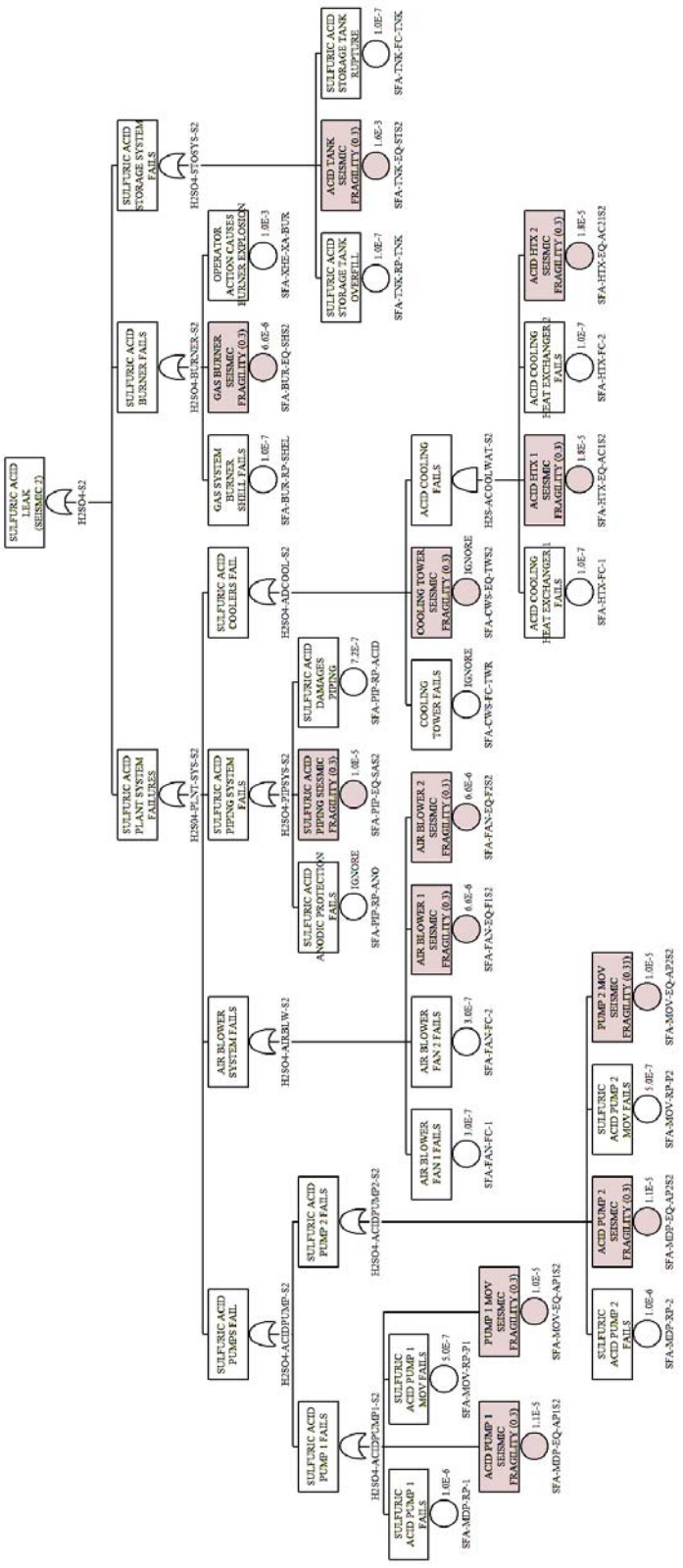

Figure 18. Fault tree H2SO4-S2.

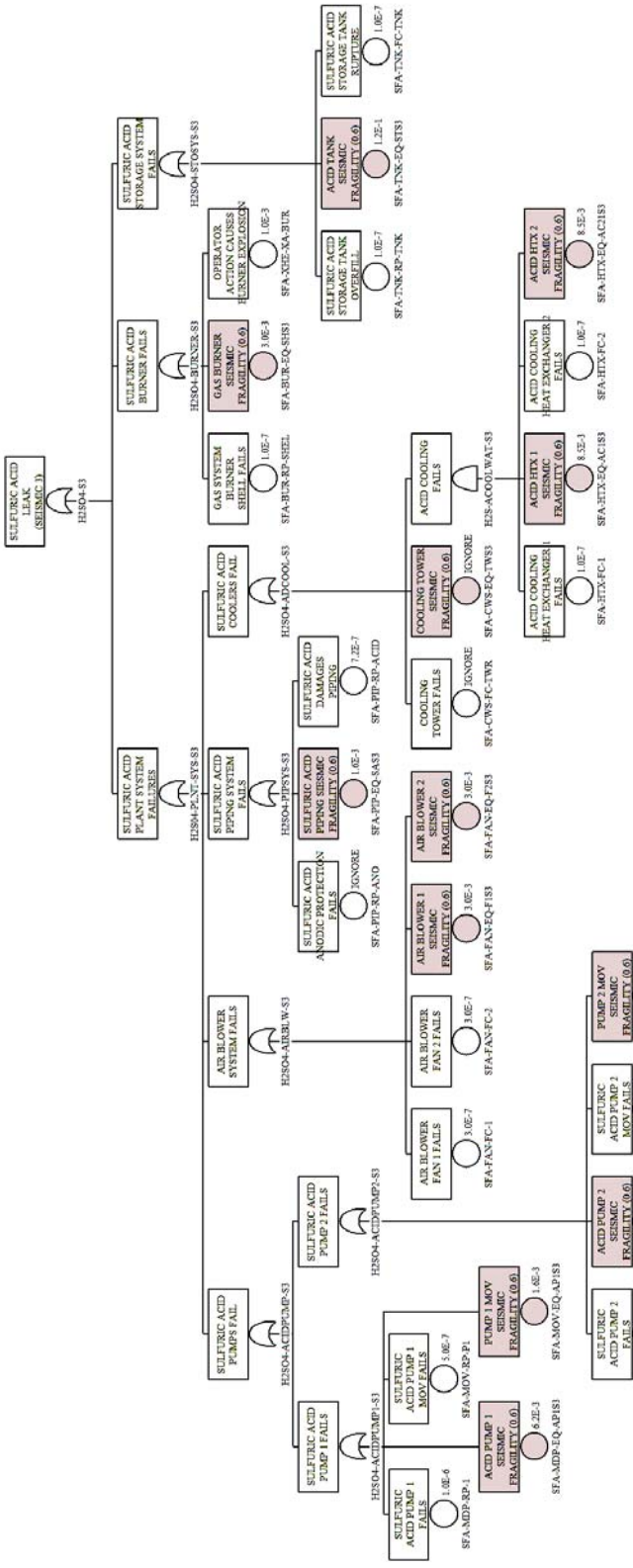

Figure 19. Fault tree H2SO4-S3.

2.3 Quantifying the Risk Model

2.3.1 Blast Analysis

To determine structural effects from detonation events, we used the blast analysis performed as part of Generic Safety Issue 106, Piping and the Use of highly Combustible Gases in Vital Areas. As part of this research, the NRC had the INL staff determine blast effects on plant structures such as reinforced concrete walls from gaseous hydrogen detonations (INEL 1994). The results of this work are documented in report EGG-SSRE-9747, *Improved Estimates of Separation Distances to Prevent Unacceptable Damage to Nuclear Power Plant Structures from Hydrogen Detonation for Gaseous Hydrogen Storage*. The blast analysis work performed by INL evaluated separation distances from hydrogen storage facilities to the current generation of light water nuclear reactor safety structures. The blast evaluation did not use equivalencing hydrogen to a TNT detonation; instead, it used a computer code that employed physics-based principles to model shock wave and the combustion process inside a reacting hydrogen mass. While the shock wave is attenuated with distance, the analysis also considered the incident wave pressure, velocity, impulse time history, and the associated reflected wave. The shock wave impulse information was used to determine the effect on the target surfaces and looked at a variety of wall conditions, including variations in wall thickness, static pressure capacity, and tensile steel factor. The calculations assumed hemispherical expansion, which should be suitable for our analysis since we are not modeling hydrogen detonations that occur significantly above ground level. For further information on the general properties of hydrogen, see Cadwallader and Herring (1999).

One of the main analysis results from the EGG-SSRE-9747 report was a determination of minimal separation distances from the point of the hydrogen detonation as a function of the detonating volume of hydrogen. This calculation was performed for different wall sizes and compositions, ranging from 12 to 24-in. wall thicknesses and static pressure capacities from 1.5 to 4.5 psi. We have taken these calculations and used the variation in structural makeup to provide a mechanism to reflect the modeling uncertainty we have on the ultimate design of major structural systems such as the HTGR containment building. The result of this modeling is shown in Figure 20. To iterate, the calculations used to construct the curves shown in this figure do not use TNT equivalence approximations.

The blast versus separation distance curves give us one portion of the risk model, that of imposing a structural load on the reactor building as a function of separation distance. For situations where we are in the lower-right regions of Figure 20, the reactor structure is assumed to have failed (we have exceeded the wall capacity, or the impulse from the detonation is greater than the wall can bear). Consequently, we need to model both the detonating volume and the separation distance in the risk model in order to determine if we are below the detonation distance curve. However, one complication here is that there is model uncertainty in the detonation distance curve, as shown in Figure 20. In this figure, we see that the uncertainty increases as the separation distance or detonating volume increases. That is, the region between the upper and lower bound curves increases, indicating that our state of knowledge becomes less precise as either the separation distance increases (with a corresponding *lower* structural wall failure probability) or as the detonating volume of hydrogen increases (with a corresponding *higher* structural wall failure probability).

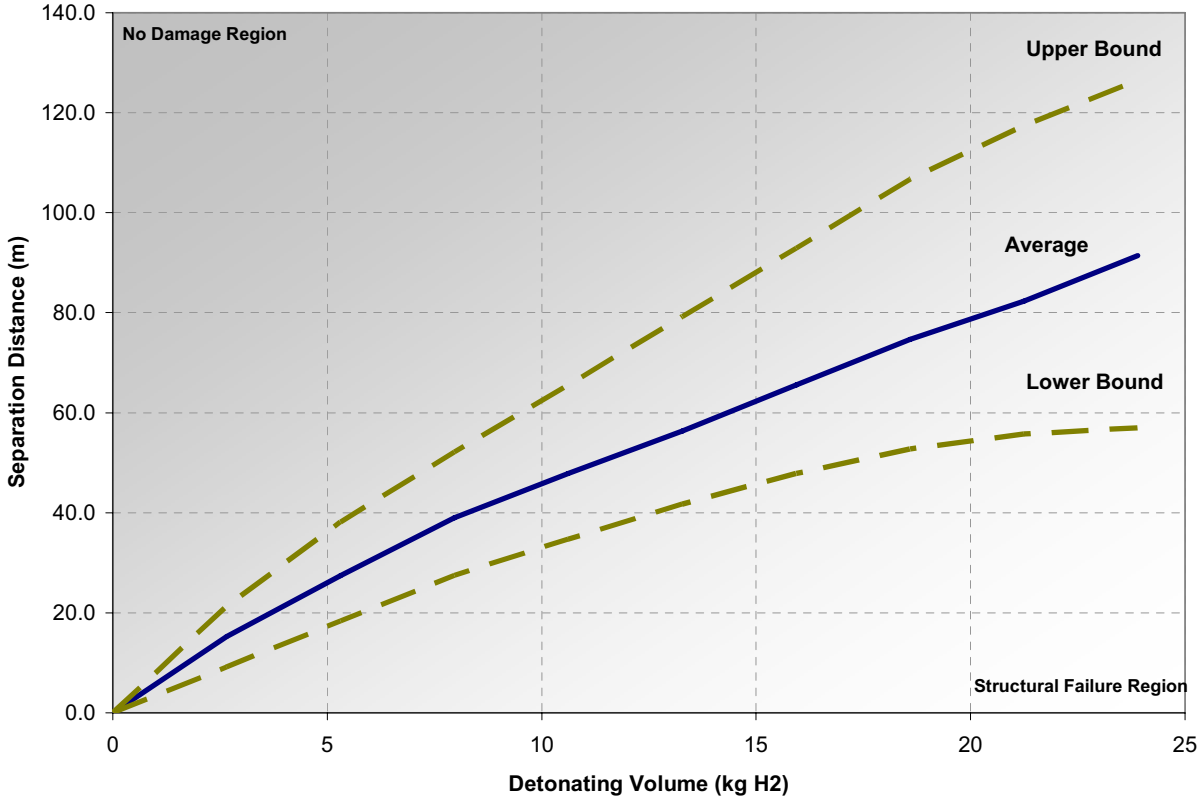

Figure 20. Separation distance as a function of the detonating volume of gaseous hydrogen, including the uncertainty in wall composition and thickness.

In order to model the detonation distance curve and its impact on the failure potential of the reactor structures, we need a calculation to find the probability that we are in the structural failure region (lower right) of Figure 20. First, we will use a standard "load-capacity" type of calculation. For this calculation, we define the following relationships:

L = the load, which is a function of V_D and the separation distance

C = the capacity

= the curves shown in Figure 20 at a particular value of V_D

where

V_D = the detonating volume

= f V.

where

f = yield (the fraction of the hydrogen that participates in the detonation

V = total volume of hydrogen in storage.

Typical values of f range from 0.05 to 0.25 (INEL 1994). For our analysis, we modeled the yield by way of a uniform distribution from 0.05 to 0.25.

f → Uniform(0.05, 0.25)

where → implies the assignment of an uncertainty distribution to the f parameter.

The capacity term represents the ability of the nuclear structure to withstand the imposed loading conditions from the detonating hydrogen. However, this capacity is uncertain, where the uncertainty depends on a specific plant configuration (e.g., distance, wall thickness, amount of hydrogen). In order to illustrate the "capacity" information, we have discretized the detonation distance curves into three cases:

C_1 → Normal(3.5 kg, 0.9 kg)

C_2 → Normal(8.0 kg, 2 kg)

C_3 → Normal(14.5 kg, 3.9 kg).

where the three cases correspond to a separation distance of 20, 40, and 60 m, respectively.

The parameterization of these distributions defined above for the capacity model represents the mean and standard deviation. For example, the second term, C_2, is the capacity defined at a separation distance of 40 m, where the mean is 8.0 kg of hydrogen, and the standard deviation is 2 kg. This response to hydrogen detonation event distributions is illustrated in Figure 21.

To determine whether the nuclear plant structures will be damaged given a hydrogen explosion, we need to calculate the probability that the load (L) is greater than the capacity (C), or

$P(L > C)$ = structure failure probability .

To perform this calculation, we used Monte Carlo simulation using 1,000,000 iterations, where at each iteration we randomly sample from the capacity and load functions and determine whether the load is greater than the capacity. The ratio of the number of times the load exceeds the structural capacity provides the structural failure probability. This calculation is repeated for the separation distances shown in Table 1, where for each distance we also list the lower bound, mean, upper bound, and standard deviation on the capacity curves shown in Figure 21.

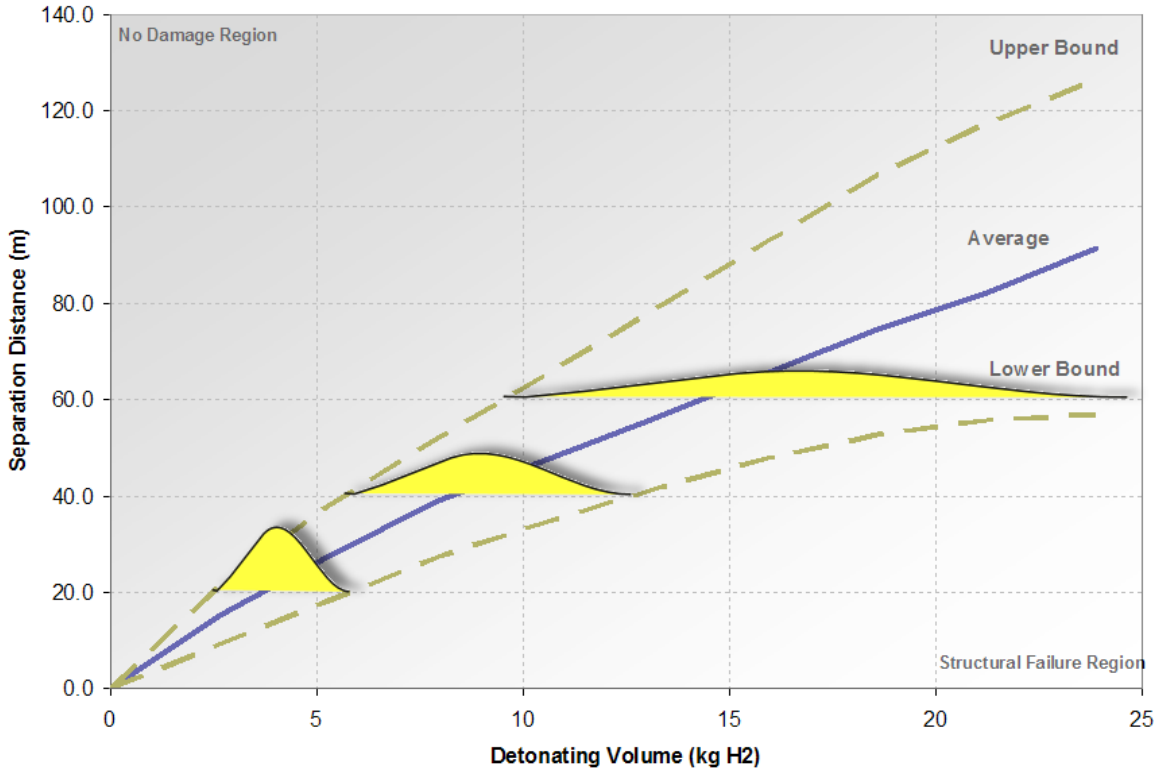

Figure 21. The structural capacity curves for the four cases used in the risk analysis model.

Table 1. Variation in structural response as a function of separation distance for hydrogen detonation events.

Separation Distance (m)	Lower Bound Volume (kg H_2)	Mean Volume (kg H_2)	Upper Bound Volume (kg H_2)	Standard Deviation Volume (kg H_2)
0	0	0	0	0.0
10	3	1.5	1	0.4
20	6	3.5	2.5	0.9
30	9	6	4	1.3
40	13	8	5.5	1.9
50	17	11	7.5	2.4
60	25	14.5	9.5	3.9
70	25	17	11.5	3.4
80	25	20	13.5	2.9
90	25	23.5	15.5	2.4
100	25	25	17	2.0
110	25	25	19.5	1.4
120	25	25	22	0.8
130	25	25	25	0.0

The load term represents the pressure impulse condition imposed upon the nuclear-related building structures. Like the capacity term, the loading is also uncertain and is a function of distance, yield, and total volume of hydrogen available. To represent the variety of these conditions, we have assumed that the largest amount of hydrogen stored at the chemical facility and available to participate in a single detonation event is 100 kg. In the worst-case scenario, the detonation will have the highest yield fraction (f = 0.25) and will result in the largest modeled detonating volume of hydrogen (V_D = 25 kg). While the INEL analysis on hydrogen combustion did not evaluate larger detonation events, it did note that "greater detonating volumes … may error on the unconservative side" (INEL 1994). However, we *do* recommend that as hydrogen is produced, it is piped offsite for compression and storage, thereby limiting the overall on-site storage in a single location to less than 100 kg of hydrogen.

Since the parameter f is uncertain and has been assigned as a uniform distribution from a yield of 0.05 to 0.25, this implies that the detonating volume for an event will also be uncertain. Further, we do not know the ultimate on-site storage capacity (assumed to be less than 100 kg) of hydrogen as it is produced, which indicates that the total volume parameter is also uncertain. The stored volume of hydrogen may range from very small amounts during initial starting conditions to full storage capacity (100 kg). Consequently, we have chosen to model this parameter with an uncertainty distribution also, specifically a uniform distribution ranging from a lower bound of 0 kg to an upper bound of 100 kg. Thus, the relationship for the detonating volume under accident conditions is:

$$V_D = f\,V \rightarrow \text{Uniform}(0.05, 0.25) \times \text{Uniform}(0\ \text{kg}, 100\ \text{kg}) .$$

If the load, L, is greater than the capacity, C, then the reactor structure is assumed to be failed. The Monte Carlo simulation using 1,000,000 iterations provides the number of times, out of the total number of iterations, that the structure will fail. The result of this calculation is shown in

Figure 22 as a function of the separation distance. In Section 3, we used this probability-of-failure curve to evaluate the sensitivity of the risk results when the hydrogen facility is moved nearer or farther from the nuclear plant.

We should note that beyond separation distances greater than 140 meters, the probability of experiencing structural damage becomes quite low (less than 1E-3). This observation is due to two reasons: the attenuation of the blast shock wave rapidly decreases with distance, and we limited the maximum on-site storage of hydrogen to 100 kg. If the total amount of hydrogen available to participate in a detonation exceeded the 100 kg value, then the structural failure probability would increase somewhat. However, as the amount of hydrogen increases, the yield fraction has a tendency to decrease, thereby neutralizing some of the impact caused by the larger amount of hydrogen.

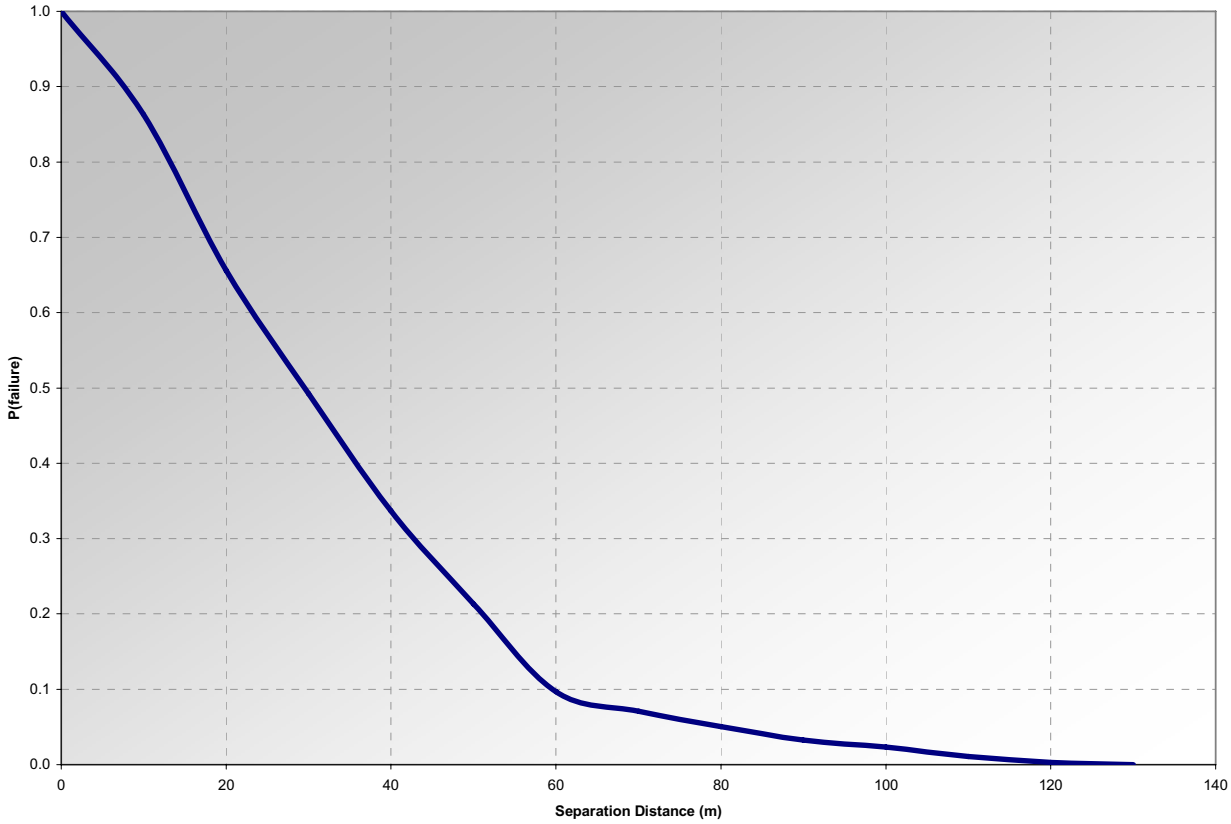

Figure 22. Probability of nuclear plant structural failure from hydrogen detonation events as a function of the separation distance between the plant and the hydrogen generation facility.

2.3.2 Likelihood of Hydrogen Events

In our risk models described in Section 2.2, we have three parameters relating to hydrogen explosions. The first is the frequency of hydrogen explosions due to "random" component failures in the hydrogen production and storage facility. The second is the probability of having an explosion when a leak of hydrogen is present. The third is the frequency of hydrogen leaks caused by human error.

To model the frequency or probability of hydrogen-related events, we reviewed past operational events at nuclear power plants. A search was conducted to identify hydrogen events at U.S. nuclear power plants that resulted in fires, leaks, explosions, or potentially explosive concentrations. The primary source of information found was the analysis performed for GSI-106 (Simion et al. 1993). Based on

information obtained from this study, it appears that several hydrogen events occurred in the United States. The review of these hydrogen events determined that 83 were a hydrogen leak, explosion, fire, or potentially dangerous concentration in the plant (see Figure 23). Of the 83 hydrogen events, 33 resulted in either a fire and/or explosion and are applicable to events that may occur at the hydrogen production facility. These 33 events are shown in Table 2.

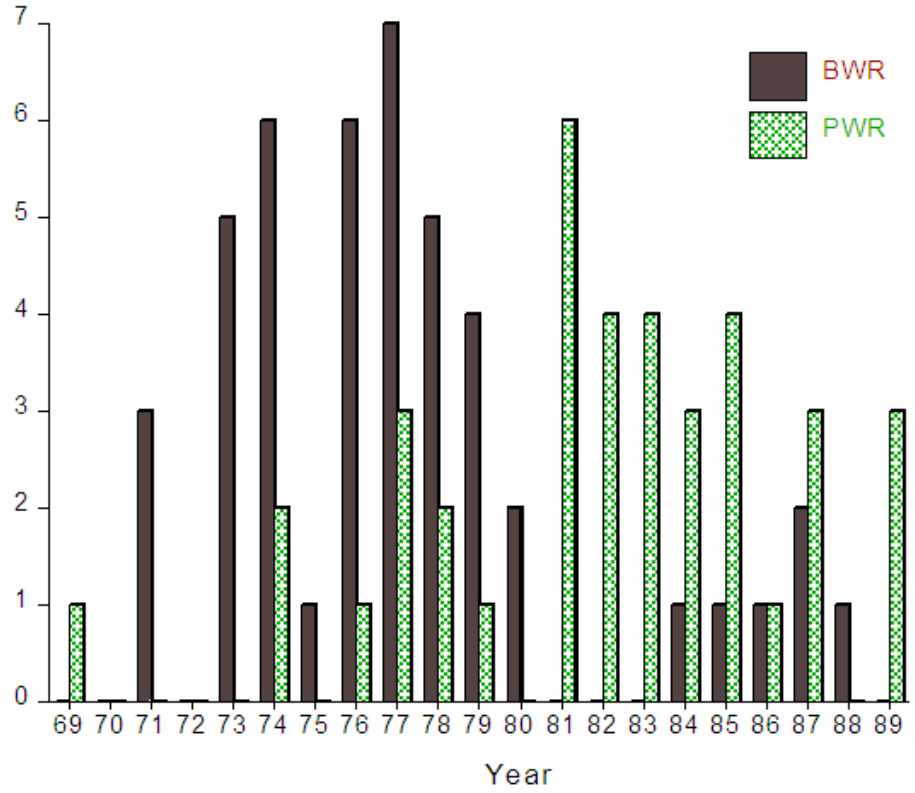

Figure 23. Hydrogen events at nuclear power plants in the United States over a 21-year period (Simion et al. 1993).

Table 2. Summary of the hydrogen events by BWR and PWR plant location, excluding technical specification concentration limits (Simion et al. 1993).

Event Location (time in reactor years)	Explosion Events	Fire Events	Uncombusted Leak	Total Events
Turbine building (1424 BWR/PWR)	2	7	7	16
Volume control tank (917 PWR)	0	0	11	11
Rad-waste gas system (917 PWR)	1	0	1	2
Hydrogen storage system (1424 BWR/PWR)	2	1	0	3
Batteries (1424 BWR/PWRs)	1	0	0	1
Total hydrogen events	6	8	19	33

The events identified in Table 2 may be used to determine applicable initiating event frequencies. Once these frequencies are identified, the accident sequences resulting from the upset condition are modeled using event trees and fault trees in the SAPHIRE software (see Section 2.2). The event trees provide the unifying structure for the hydrogen risk PSA. An event tree is a graphical description of accident sequences, where the first branch on the event tree is the initiating event. It then proceeds from left to right, calling out and showing how the systems (top events) are reacting to the initiating event (on the left). The sequence analysis proceeds through this path, with the success or failure of a top event at each branch of the tree.

First, to evaluate the frequency of "random" hydrogen explosions, we used the events that led to explosions (a total of six events) in the nuclear plant experience. We used a Jeffreys noninformative prior distribution, which was updated with the event data using standard Bayesian practices (Atwood et al. 2003). Thus, for each applicable plant area, we end up with a gamma distribution with parameters $\alpha = x + 0.5$ and $\beta = t$, where x represents the number of explosion events and t represents the data collection time (in reactor years). The result of this analysis is the curves shown in Figure 24.

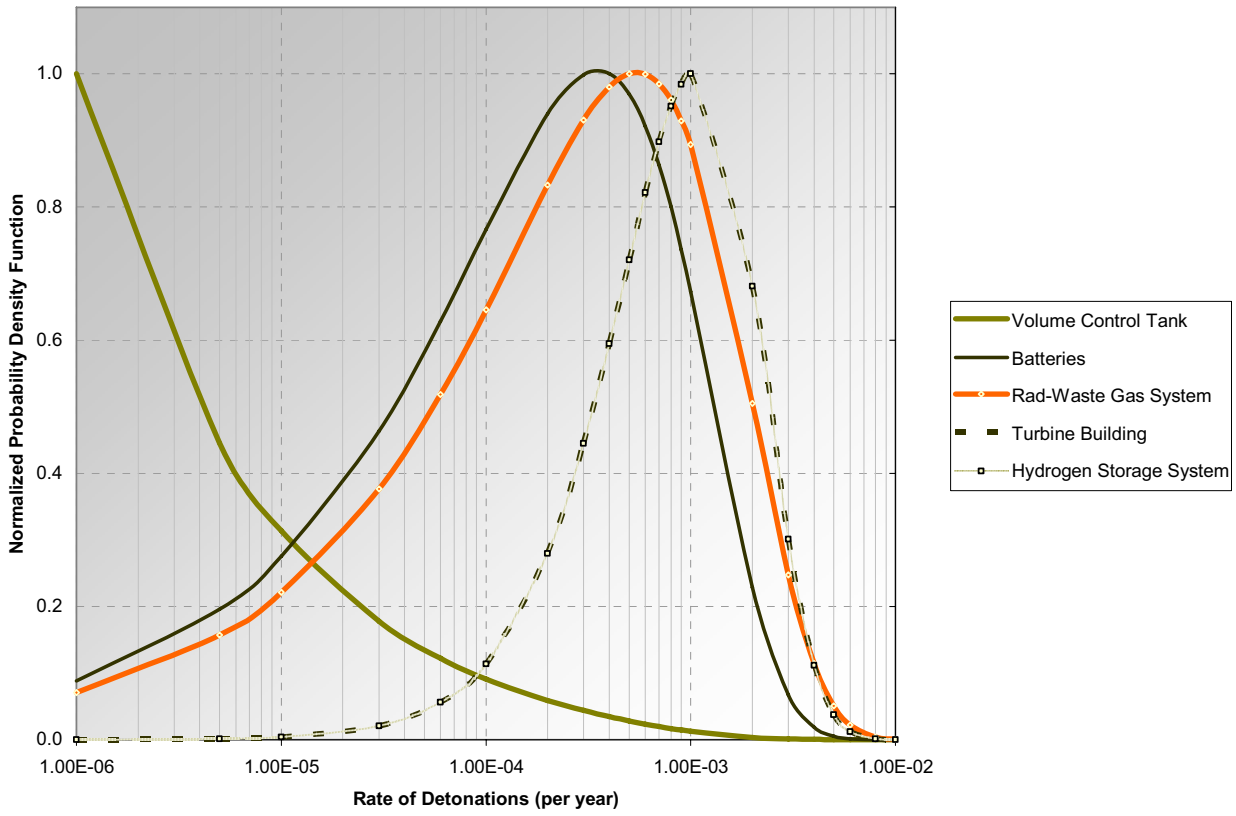

Figure 24. Hydrogen detonation event rate of occurrence for five plant areas.

Since each of the curves shown in Figure 24 provides information potentially relevant to the rate of seeing hydrogen detonation events in the chemical facility, we use this information as part of the overall uncertainty on this rate parameter. In other words, the rate of hydrogen detonation events, λ_{HD}, is provided by factoring each of the curves above. If we believed one area were more representative of the operation of the chemical facility with respect to the processing and storage of hydrogen, we could weight that area's curve higher. However, we weighted the area information equally, which then provided the uncertainty information for the total rate, as shown in Figure 25. This distribution was used in the

SAPHIRE risk analysis model and was entered as a probabilistic histogram. Note that the mean of this detonation rate is 8E-4/yr, which indicates that the mean frequency of "random" hydrogen explosions is about 8E-4/yr. This initiating event rate is comparable to the frequency of experiencing a complete loss of service water or having a small-break loss-of-coolant accident in the current fleet of nuclear power reactors.

The second parameter of interest for hydrogen events was the probability of having an explosion if a leak of hydrogen were present. For this probability, denoted as P(explosion | leak), we again used the data in Table 2. However, we now focus solely on the conditional probability that given we see an event, what is the likelihood that the event results in a hydrogen explosion. For this analysis, we grouped the explosion and leak event data into a single category, which gave us a total of 25 events. Of these 25 events, a total of six resulted in explosions. Thus, for our data analysis, we have 25 events and 6 "failures" (those which go to an explosion). As for the frequency of hydrogen explosions, we used a Jeffreys noninformative prior distribution, which was updated with the event data using standard Bayesian practices (Atwood et al. 2003). In this case, then, we end up with a beta distribution with parameters $\alpha = x + 0.5$ and $\beta = 0.5 + n - x$, where x represents the number of explosion events (6) and n represents the total number of events (25). The result of this analysis is the cumulative probability curve shown in Figure 26. This probability distribution was used in the SAPHIRE risk analysis model. The mean of this distribution is a probability of 0.25, or about 25% of the hydrogen leaks seen in the operational record resulted in an explosion.

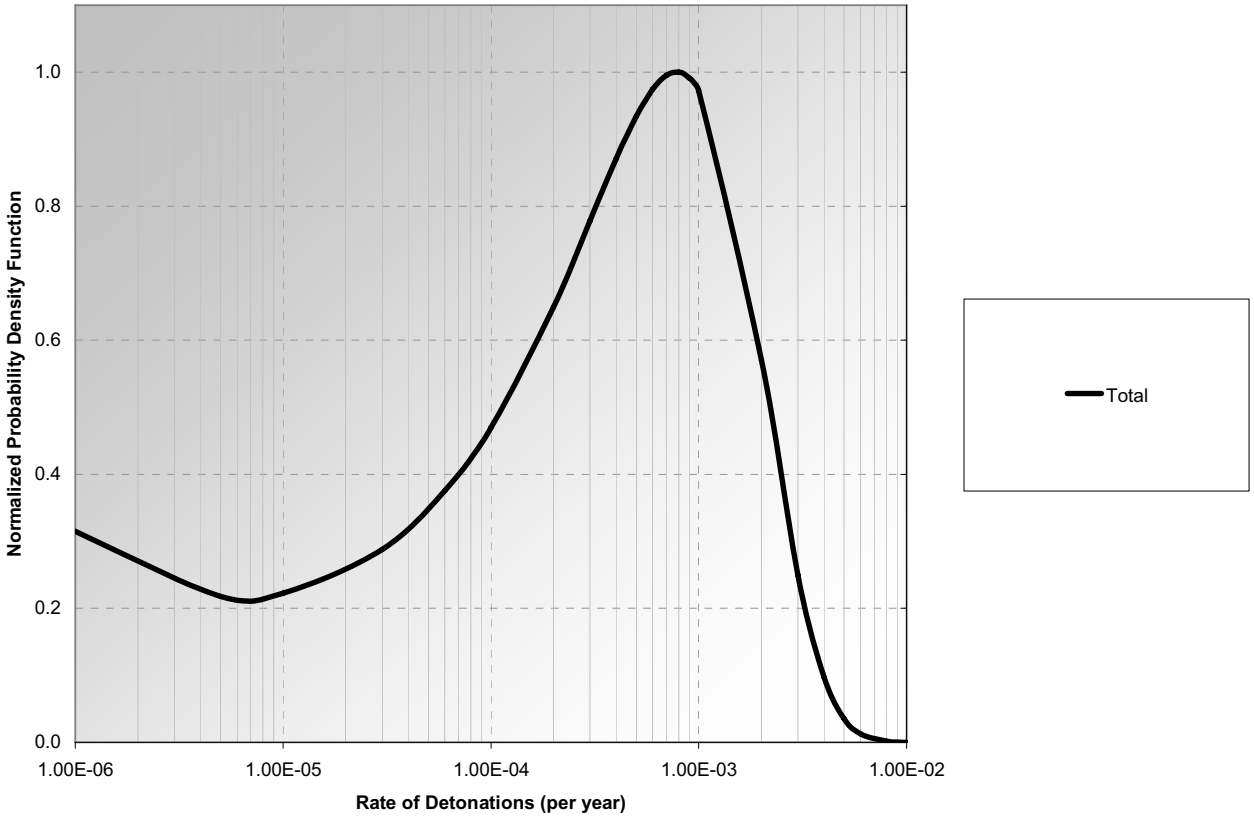

Figure 25. The uncertainty on the total rate of hydrogen detonation events applicable to the chemical facility.

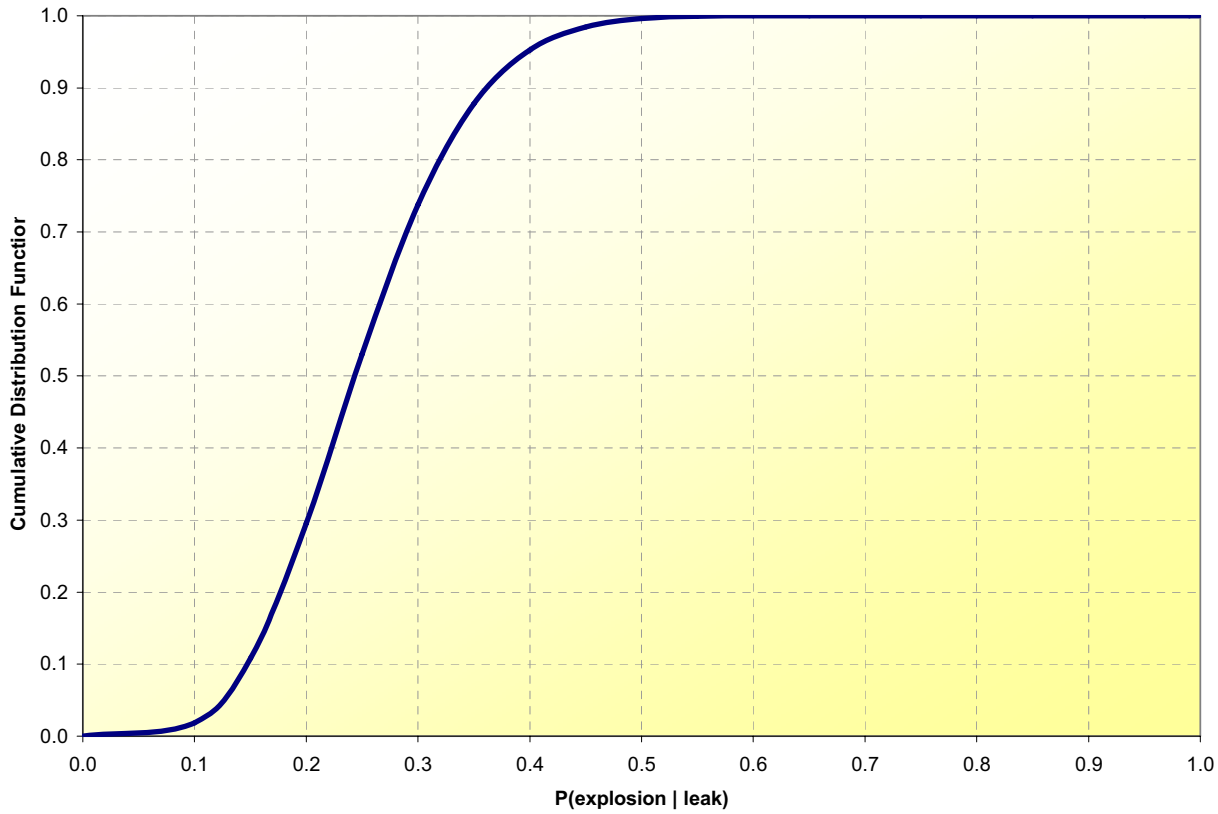

Figure 26. The conditional probability of seeing an explosion given that there is a leak of hydrogen.

The third parameter of interest for hydrogen events was the frequency of hydrogen leaks caused by human errors. To determination this event, we used information in the GSI-106 report (Simion et al. 1993). In this report, the hydrogen events for the BWR off-gas system had identified several events that had causal mechanisms relating to human activities on that system. For example, one event was caused by changes in maintenance procedures. A second event was due to "personnel error." These operation events spanned a collective time of 507 years of plant operation. Within this time, a total of five human-related events were identified. As for the frequency of hydrogen explosions, we used a Jeffreys noninformative prior distribution, which was updated with the event data using standard Bayesian practices (Atwood et al. 2003). In this case, we end up with a gamma distribution with parameters $\alpha = x + 0.5$ and $\beta = 0.5 + t$, where x represents the number of human-related hydrogen events (5) and t represents the total time (507 years). The result of this analysis is the cumulative probability curve shown in Figure 27. This distribution was used in the SAPHIRE risk analysis model. The mean of this distribution is a rate of 0.01 per year, which is comparable to the rate of seeing a loss-of-offsite power in the current fleet of nuclear power reactors. We should note that this rate is only the frequency of seeing hydrogen leaks—the probability of having an explosion given a leak is also factored into the accident sequence via the event tree model described in Section 2.2.

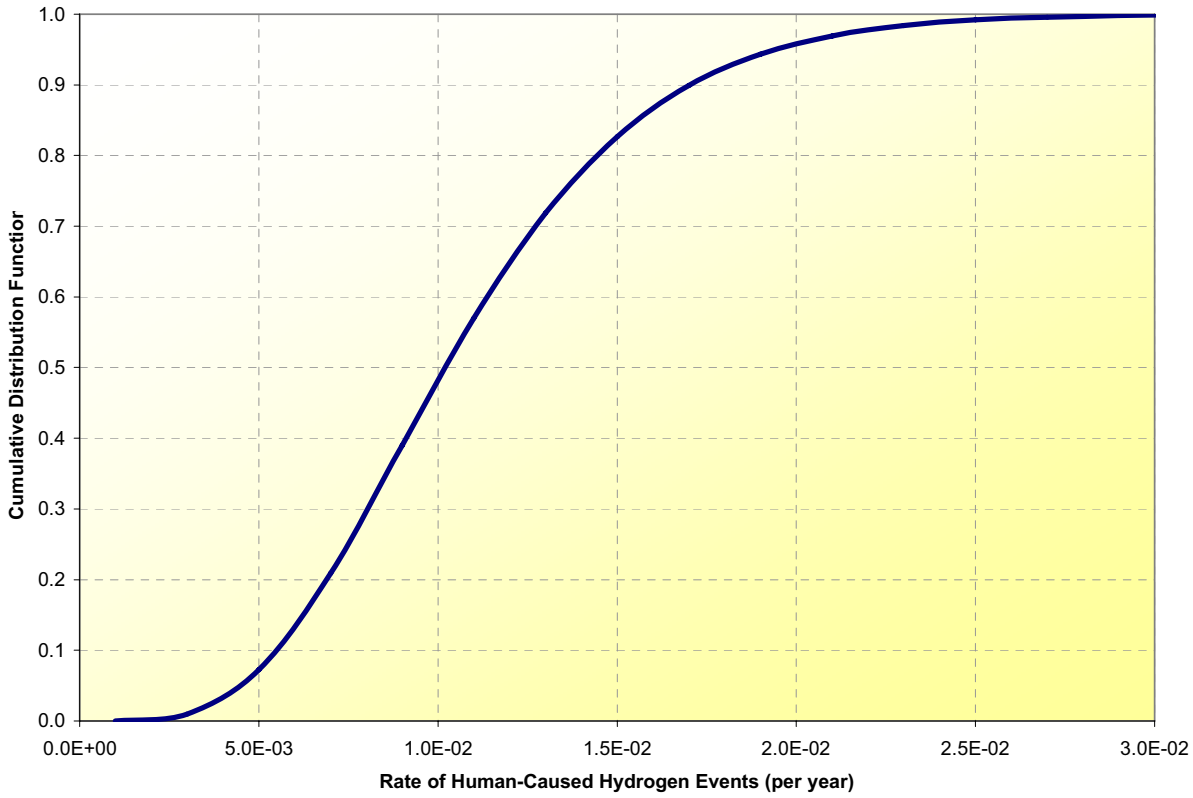

Figure 27. The rate of human-caused hydrogen events (per year).

2.3.3 Probability of Causing Core Damage

For some of the accident scenarios described in Section 2.2, we require a probability that the scenario (to that point) will eventually lead to a core damage event. In our model, we represent two distinct conditions, first where we require probability of core damage given that a hydrogen detonation physically damages the nuclear plant structures. For the analysis in this report, we did not have detailed models relating to consequential core damage following structural damage of the nuclear plant building. Second, for those situations that do not damage the nuclear plant structures, we still have an upset condition (e.g., potential loss-of-offsite power, loss of the secondary heat-exchanger, plant transient) that requires knowledge of the probability of core damage. To evaluate these two conditions, we ideally would have access to a full-power PSA for the HTGR. However, such a study does not exist for the current plant design, which resulted in using a PSA performed in the mid-1980s by GA Technologies.

The PSA performed by GA Technologies (Everline 1984) evaluated a 558-MW(e) modularized prismatic HTGR. This design is similar to the point design (MacDonald et al. 2003) currently being evaluated for the hydrogen-producing VHTGR (very-high-temperature gas reactor). In the GA Technologies PSA, the reactor building contains a reactor module embedded in the earth and consists of a concrete enclosure encasing the reactor internals. The reactor building also functions as a filtration system to capture particulates and halogens. However, this system is not able to withstand accident loading conditions. For our analysis, we assume that a loading of 7 kPa or more will result in functional failure of portions of the aboveground portion of the reactor building. Note that 7 kPa is the peak pressure identified in RG 1.91 for wind loading of nuclear-related structures, systems, and components

(SSC).[a] Of further concern for scenarios that compromise the rector building is the potential for air ingress if the reactor's primary coolant boundary is damaged.

One scenario of concern is a detonation or deflagration event that impacts the aboveground reactor building structure but does not cause damage to the reactor's primary coolant boundary nor result in a reactivity insertion event. This type of event will most likely result in (at best) a plant transient. However, the 1984 PSA indicates, for these types of events, that "… releases with an intact primary coolant boundary and a reactor building failure have no identifiable safety risk significance."

The frequency of core damage is our primary metric for evaluation in this report, not the release frequency or associated risk. Thus, from the 1984 PSA, we have extracted potentially applicable risk scenarios and their quantification insights. This information is shown in Table 3. While the 1984 PSA did quantify uncertainty on release consequences, the uncertainty for the scenarios identified in Table 3 were not provided—however, we expect the sequence level uncertainty to be typical of that found in more recent PSAs, or about a factor of 10 from the specified frequency to both the 5^{th} and 95^{th} percentiles. Further, we assume that the values supplied represent means; we have entered the values as means in the SAPHIRE risk model.

We used the information in Table 3 to represent the conditional probability of core damage from a variety of accident scenarios. For example, in the seismic event tree for a 0.2 g earthquake (see Section 2.2), Sequence 2 represented the earthquake, no failure of the hydrogen production facility, but core damage due to the earthquake itself. In this sequence, the probability of core damage given the earthquake is found in Table 3, third column, and has a value of 6E-9 per earthquake event.

Table 3. Accident scenario information for the GA Technologies 1984 HTGR PSA.

Type of Accident Scenario	Frequency (per year)	Conditional Probability of Core Damage	Comments
Loss of offsite power and subsequent trip	1E-9	1E-7	Active cooling in this scenario was dependent on function of on-site diesel generators. The passive cooling does not require diesel generators.
Loss of the Heat Transport System	5E-8	2E-8	Active cooling in this scenario was dependent on available ac-power sources.
Anticipated transient without scram	6E-7	2E-7	The probability of failing to scram the reactor was assumed to be 1E-5 per demand, which is similar to that found for the current fleet of light water reactors in the U.S. [US NRC, 2001b]
Earthquake (0.2 g)	2E-11	6E-9	Earthquake frequencies were taken from the Zion and Seabrook PRAs. It is assumed that seismic events also result in a loss of offsite power.
Earthquake (0.4 g)	1E-11	2E-8	The reactor Heat Transport System is assumed to be failed for g levels at 0.4 g and above.
Earthquake (>0.4g)	2E-11	2E-7	The reactor Heat Transport System is assumed to be failed for g levels at 0.4 g and above.

For the case where a hydrogen detonation results in structural damage to the nuclear facility, we needed to determine the probability that cooling systems remaining in the plant will be operational. For this scenario, we used the 1984 PSA, specifically for the dominant seismic sequences that question the availability of cooling following an earthquake. While the response of equipment during an earthquake is

[a] Note that a design basis tornado with wind speeds of about 195 mph could have a pressure loading of about 10 kPa.

different than that of building structural damage, this scenario was the closest match to the hydrogen detonation event in our PSA. From the 1984 PSA, the relevant cooling system failure probabilities were:

P(core damage | structural damage) = P(heat transport fails) * P(shutdown cooling fails)
= 1.0 * 2E-2
= 2E-2 .

2.3.4 Seismic Events

The frequency of earthquake events is from Lawrence Livermore report NUREG-1488 (NRC 1993), which is an update of the earlier report NUREG/CR-5250 (Bernreuter et al. 1988). Since we decomposed the earthquake sizes into three groups, we had to evaluate the frequency for each group to determine the group's average rate. The results of this analysis are shown in Table 4. The last column in this table presents the initiating event frequencies used in the SAPHIRE risk model for the seismic event trees.

Table 4. Earthquake frequency information for the three seismic event trees.

Ground Acceleration (g)	Exceedance Frequency (1/y)	Seismic Bin for Model	Seismic Bin Acceleration (g) (geometric average)	Seismic Bin Frequency (1/rcry)
0.050	9.59E-04	1	0.100	8.33E-04
0.075	5.43E-04	(0.05 -0.2g)		
0.150	1.81E-04			
0.200	1.26E-04			
0.250	7.10E-05			
0.300	4.94E-05			
0.400	2.69E-05	2	0.283	9.91E-05
0.500	1.62E-05	(0.2 - 0.4g)		
0.650	8.60E-06	3	0.632	2.41E-05
0.800	5.05E-06	(0.4 - 1.0g)		
1.000	2.76E-06			

Once the frequency of the particular size of earthquake is known, we must then consider the failure probability for components in the hydrogen production facility conditional upon the size of the earthquake. The "size" of an earthquake is measured by its peak ground acceleration, or g. The fragility of a component is the probability that the component fails, which is a function of g (the larger the earthquake, the more likely the component is to fail). For a particular component, we let A_f denote the *failure acceleration*, or the ground acceleration that is just sufficient to cause a component to fail:

$$A_f = \alpha \varepsilon_R \quad .$$

where α is the median failure acceleration capacity of the component and ε_R is a lognormal variable representing the random variation on the component failure. The ε_R term is a function of two parameters, $Beta_R$ and $Beta_U$, which have been tabulated for a variety of components. For our analysis, we used component fragility information from NUREG/CR-6544 (Budnitz et al. 1998). The functional form of the fragility equation to determine the failure probability as a function of the g level is given by:

$$A_f = \Phi\left(\frac{\ln(g/\alpha)}{\sqrt{Beta_R^{\,2} + Beta_U^{\,2}}}\right)$$

where Φ is the standard normal cumulative distribution function. We used this equation to determine the probability that a component will fail given that is has been subjected to an earthquake of size g. The

results of the analysis are shown in Table 5, where the component data have been gathered from NUREG/CR-6544 (Budnitz et al. 1998). These failure probabilities are used in the SAPHIRE risk model for the respective fault tree logic models relating to the seismic event trees.

Table 5. Probability of failure for a variety of component types due to earthquakes of three different acceleration (g) levels.

Component Description	Median Capacity (g)	$Beta_R$	$Beta_U$	Component Seismic Fragility First g group Second g group Third g group
Accumulator	2.5	0.3	0.35	1.45E-12 1.14E-06 1.43E-03
Air Handling Unit	2.5	0.3	0.4	6.06E-11 6.59E-06 2.98E-03
Buried welded steel piping	2	0.25	0.3	8.51E-15 2.76E-07 1.59E-03
Cable Trays	2.5	0.35	0.5	6.67E-08 1.79E-04 1.21E-02
Heat Exchanger	1.9	0.3	0.35	8.44E-11 1.81E-05 8.47E-03
HVAC duct	2.5	0.35	0.5	6.67E-08 1.79E-04 1.21E-02
Large storage tanks	1.1	0.3	0.35	9.87E-08 1.61E-03 1.15E-01
Motor pumps	2	0.3	0.35	4.05E-11 1.11E-05 6.23E-03
MOVs	3.8	0.35	0.5	1.26E-09 1.04E-05 1.65E-03
Piping	3.8	0.35	0.5	1.26E-09 1.04E-05 1.65E-03
Pressurizer	2.5	0.3	0.4	6.06E-11 6.59E-06 2.98E-03
Safety or check valves	3.8	0.35	0.5	1.26E-09 1.04E-05 1.65E-03
Switchgear	3.1	0.3	0.35	4.69E-14 1.04E-07 2.81E-04
Transformer	1.9	0.3	0.35	8.44E-11 1.81E-05 8.47E-03

2.3.5 Parameters from the SAPHIRE Risk Model

Within a PSA, the fault trees and event trees provide the logic modeling structure. However, within this structure, the basic events (i.e., those events at the lowest level modeled in the fault trees) provide the probabilistic information needed to determine results such as core damage frequency. The discussion in Section 2.2 overviews the models used for our analysis. In Appendix B, we list all of the basic events used in the SAPHIRE risk model. In the appendix, we provide information on the basic event, including the event name, probability (or frequency, per year, if it is an initiating event), uncertainty distribution type and parameters, description, 95th percentile value of the uncertainty distribution, and supplementary notes, as illustrated below.

EXPL-H2LEAK	**2.50E-01**	Beta(6.5, 19.5)
EXPLOSION DUE TO HYDROGEN LEAK		95th = 3.98E-1
Note:		

2.3.6 Nominal Risk Results

The SAPHIRE risk model developed using the parameters and structure described in this section provide for three distinct end states or accident scenarios. First, we have modeled situations that lead to damage of the reactor core—labeled "core damage." Second, hydrogen detonation events may damage the reactor structures but do not lead to core damage—these scenarios were labeled "damage." The outcome of these scenarios will be plant upsets such as losses of offsite power or transients and will include dispersal of some chemical inventory, as described in Appendix A. Third, are events that lead to damage within the hydrogen production facility but do not damage the reactor core or structural facilities—these are labeled "dispersion." These events are expected to lead to a dispersion scenario, as modeled in Appendix A. Both the second and third types of scenarios lead to dispersion events—events that may impact the habitability of the reactor control room and associated facilities. Further, events in the second category are worse in the sense that in addition to the dispersion of chemicals, nuclear plant structures may be compromised (for example, the control room if located near the hydrogen production facility). However, due to the lack of detailed design information relating to the location, type, and structure of the control room, we did not model control room degradation in conjunction with these types of dispersion scenarios. Thus, we combined scenarios of the second and third categories into the general case of dispersion events (note that the accident frequency of the second category was about 10% of the third category).

For the nominal assessment, we assumed that the reactor was situated 60 m from the hydrogen production facility. This assumption implies that the primary on-site hydrogen storage subsystem is located 60 m from the nearest nuclear plant structure. Further, for the nominal case, we assume that no mitigating systems (e.g., blast wall) are used. For this nominal distance, we used the SAPHIRE risk model to quantify the mean (or expected) frequency for both core damage and dispersion events. The results of this calculation are shown in Table 6.

Table 6. Mean risk results for the hydrogen plant SAPHIRE risk model.

Case	Core Damage Frequency (per year)	Dispersion Frequency (per year)
Nominal	7E-6	3E-3

The first result in Table 6, the core damage frequency, can be compared directly with the risk metrics and guidance described earlier, for example that in used in RG 1.174. The dispersion result, however, deserves additional discussion on the interpretation. The nominal analysis yielded a frequency of dispersion events from the hydrogen facility as 3E-3 per year. This frequency should not be confused with a core damage frequency, since dispersion events do not lead *directly* to core damage. However, it is

possible that impacts to a nuclear power plant control room may lead to some events that result in loss of reactor control and subsequent core damage. Further, while we were able to quantify the frequency of such releases, the probability that a control room is compromised is a function of a variety of site-specific features, such as prevailing winds, time of year, local geography, location of the leak, and specific chemical types. Note that RG 1.174 gives no quantitative guidance on what would be an acceptable initiating event frequency for accident scenarios.

Regulatory Guide 1.78, which discusses the evaluation of control room habitability, is deterministic in the sense that it requires detailed analysis when hazardous chemical are located within 482 m of the control room (NRC 2001a). Consequently, with the separation distances we evaluate in this report, it is expected that much of the hydrogen production facility would fall under the umbrella of this Regulatory Guide. If RG 1.78 were risk-informed, then analysis related to the frequency of chemical release and the probability of impacting the control room would be applicable. However, this risk-informed aspect is not in that guide. Instead, we evaluated the consequence of a variety of chemical releases (see Appendix A), present that information with the calculated frequency of a dispersion, and note that the frequency of core damage should be significantly lower than the dispersion frequency, since the dispersion event must travel to the control room (or intake structures), bypass filtration systems, incapacitate the nuclear plant staff, and ultimately lead to loss of control, which then goes to a core damage event. Nonetheless, due to the fact that a complex chemical processing facility is going to reside next to a nuclear power plant, we examine a sensitivity case in Section 3 wherein the control portion of the plant is moved a distance from the nuclear facility.

3. DESIGN MODIFICATION ANALYSIS

While the nominal risk analysis results indicate that the frequency of core damage (7E-6/yr) is low at a separation distance of 60 m, these results are above the regulatory threshold (1E-6/yr) normally considered by the NRC in such guidance as RG 1.174 (for example, see Figure 4). Current risk-informed practices allow for plant modification when the risk increase is "very small," where the criterion of this term is:

> "When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF" (NRC 2002).

Since the risk results are on the cusp of a regulatory threshold, we undertook several sensitivity analyses, looking at a variety of methods to help mitigate—or in some cases remove—risk drivers. As part of our sensitivity analysis relating to potential design modifications, we evaluated six different situations:

1. Varying the separation distance to the maximum distance evaluated in this report
2. Constructing an earthen barrier between the nuclear and chemical facilities
3. Constructing the nuclear facility primarily underground
4. Constructing blast panels near the chemical facility to dampen overpressure events
5. Constructing the chemical facility primarily underground
6. Moving the nuclear plant control room offsite.

We describe each of these analysis cases and discuss the results of the analysis for risk and engineering insights. Note, however, that while we did perform sensitivity analyses to look at risk sensitivities, the overall analysis should be considered to be a best-estimate analysis. Since the available detailed design information is limited, the analysis models used in this report tend to go toward the conservative side when considering the uncertainty in the best-estimate calculations. Also, for these sensitivity calculations, we focused solely on risk insights and did not consider impacts relating to the cost of construction or the operation of the plant.

3.1 Case 1: Vary the Distance

For this case, we allow the distance between the HTGR and the hydrogen production facility to vary. The risk calculations are repeated as the separation distance is either increased or decreased from the initial distance of 60 m. We illustrate this case in Figure 28. For our analysis, we moved the chemical facility as close as 20 m from the reactor and, in increments of 20 m, increased the separation distance to a total of 140 m. At each distance, the SAPHIRE risk model was reevaluated and the mean risk value determined by running an uncertainty analysis.

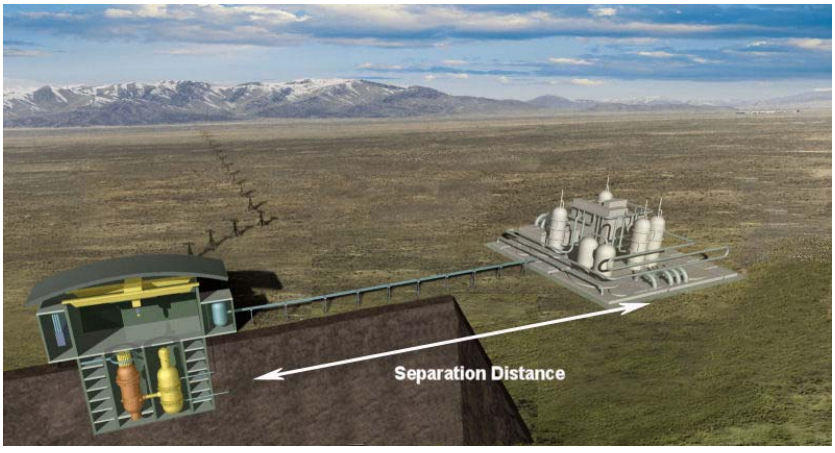

Figure 28. Case 1: varying the separation distance between the nuclear and hydrogen production facilities.

In Figure 29, we plot the results of the separation distance analysis, where we show the core damage frequency from the three initiating events we considered: seismic events, human errors, and random explosions. In this figure, we provided a demarcation to indicate the point of "very small" risk (at 1E-6/yr), where we see distances for separation above about 110 m, the risk level is above the demarcation. However, this observation should be tempered by the fact that these calculations do not consider any mitigating measures such as those described in the remainder of this section.

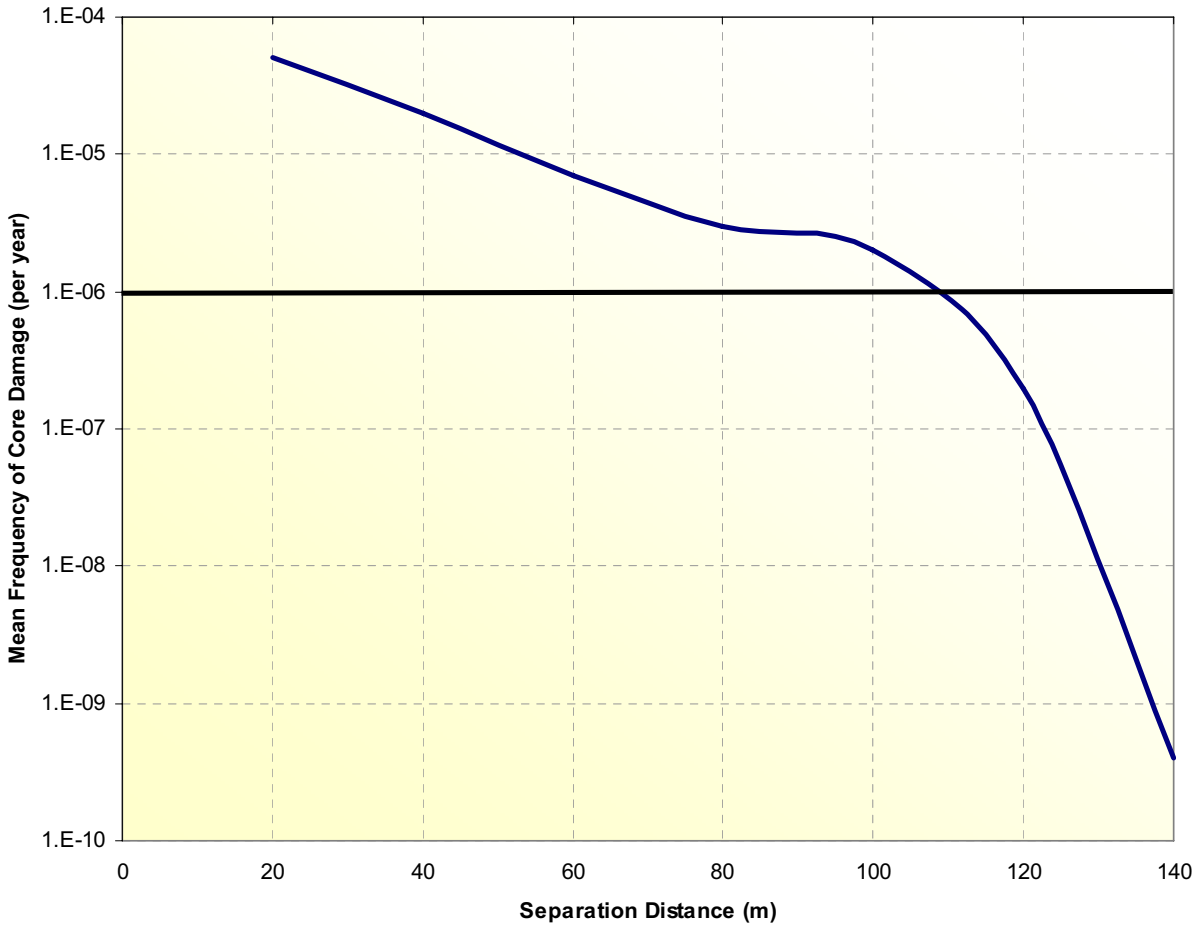

Figure 29. Core damage risk as a function of increasing the separation distance between the hydrogen production facility and the nuclear plant.

The results shown in Figure 29 follow the general curve shown where the risk level decreases slowly until reaching separation distances greater than 100 m. The driver for this behavior is that for separation distances *less* than 100 m, the risk is dominated by scenarios where hydrogen detonation events damage the reactor facility leading to structural damage and consequential core damage. As the separation distance increases, the likelihood of structural damage from a detonation drops and eventually goes to zero—however, other types of core damage scenarios then become more likely (such as the hydrogen detonation leading to a plant upset condition that then leads to core damage). The scenarios that occur for large separation distances (greater than 120 m) have very low risk, on the order of 1E-9/yr. Consequently, the two types of scenarios contribute to the overall risk at the intermediate (80 to 120 m) separation distances.

When the separation distance is increased to the maximum we evaluated (140 m), the frequency of dispersion events does not change, since it is the explosion within the chemical facility that results in the dispersion. And, at this maximum distance, the consequences of the dispersion event only changes minimally, since the dispersion footprint is measured in hundreds of meters (see Appendix A).

3.2 Case 2: Construct a Barrier between the Facilities

In order to mitigate potential detonation events that are initiated from the hydrogen facility and impact the nuclear plant, an earth barrier may be placed directly between the two facilities. This scenario is illustrated in

Figure 30. If the barrier is at a height approximately equal to the nuclear facility, blast waves from a detonation originating from the hydrogen production facility will be dampened and reflected away from the nuclear structure. To model this situation, we set the probability of structural damage from hydrogen detonation scenarios to zero. Under this assumption, the mean core damage risk decreases to 4E-10/yr, resulting in a negligible core damage frequency. However, the mean frequency of dispersion events stays at 3E-3/yr since the barrier does nothing to prevent upsets in the chemical facility. Additionally, the barrier is not expected to be high enough from the ground level to block chemical dispersion events from reaching the nuclear facility if the dispersion event occurs.

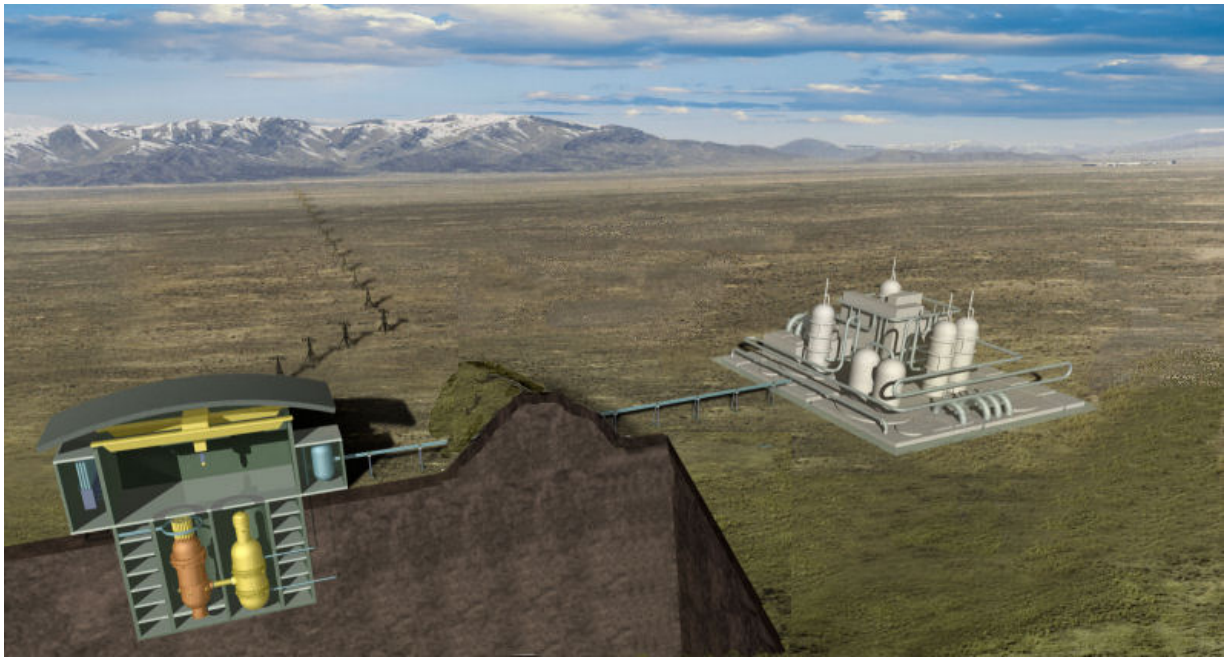

Figure 30. Case 2: Constructing an earthen barrier between the nuclear and chemical facilities.

3.3 Case 3: Construct the Nuclear Facility Underground

The current point design for the HTGR has considered and suggested that the entire reactor confinement structure be placed underground (MacDonald 2003). However, we considered accident scenarios that potentially damage aboveground structures that, if compromised, may then affect other portions of the plant located below these structures. We did not assume that a core damage condition would directly result from this type of scenario, but there was a nonnegligible probability of seeing core damage as a result of the hydrogen detonation and subsequent structural damage.

Scenario Case III considers the net reduction in risk if all major nuclear-related structural bodies were located underground, out of the reach of any possible hydrogen blast effects (see Figure 31 for an illustration). For this situation, we would be precluding the possibility of core damage from structural damage. But other risks, such as hydrogen explosions causing plant transients or losses of offsite power, are still considered.

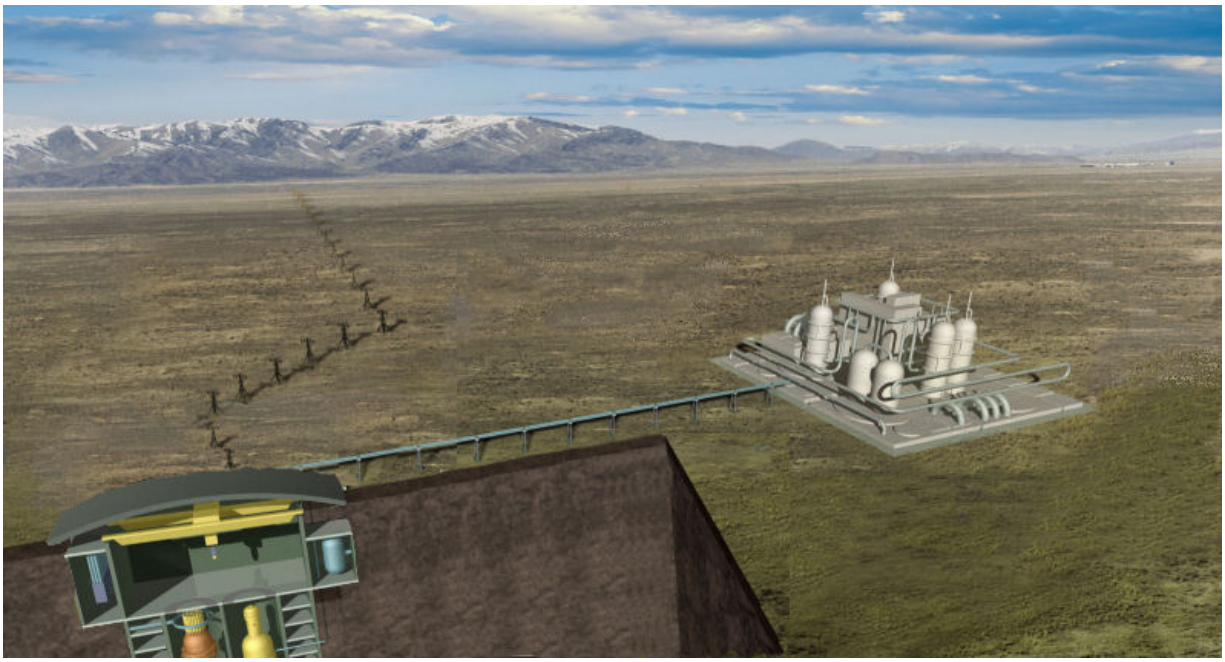

Figure 31. Case 3: Constructing the nuclear facility primarily underground.

To model the situation where the nuclear plant is underground, we set the probability of structural damage from hydrogen detonation scenarios to zero. Under this assumption, the mean core damage risk decreases to 4E-10/yr, resulting in a negligible core damage frequency. Note that we did not consider a potential reduction in dispersion risk events since the control room, while underground, would still require intake structures above ground.

3.4 Case 4: Construct Blast Panels near the Chemical Facility

Like Case II, this sensitivity case focuses on the construction of a passive system designed to mitigate the hazards associated with hydrogen detonation and deflagration events. This case, however, focuses on the construction of blast panels between the hydrogen storage and the nuclear facility (see Figure 32 for an illustration). The goal of the blast panels is to dampen the detonation shock wave—in other words, the panels are a surrogate for increasing the separation distance. To model this situation, we set the probability of structural damage from hydrogen detonation scenarios to zero. Under this assumption, the mean core damage risk is 4E-10/yr. Note that for this analysis, we assumed that the panels themselves would not become airborne missiles (for example, by permanently attaching one side of the panel to a fixed structure) or, if the panel does become airborne, the energy imparted to the panel will not be sufficient to cause enough structural damage to the nuclear facility to result in core damage.

Again, the mean frequency of dispersion events stays at 3E-3/yr, since the barrier does little to prevent upsets in the chemical facility.

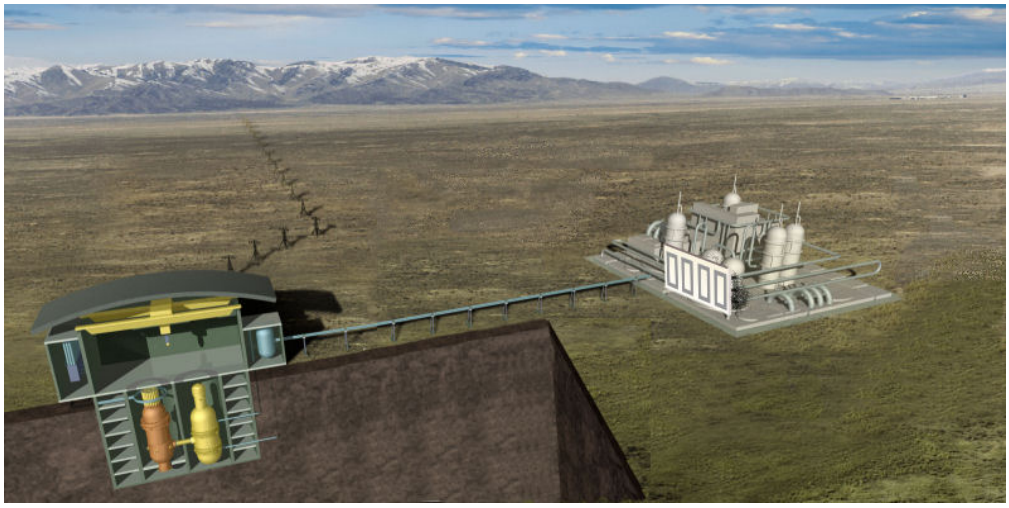

Figure 32. Case 4: Constructing blast panels near the chemical facility to dampen overpressure events.

3.5 Case 5: Construct the Chemical Facility Underground

In this sensitivity case, we propose to construct critical portions of the chemical facility underground (see Figure 33). The scenario considers the net reduction in both core damage risk (the shock wave from a hydrogen detonation would be dampened by the ground) and the potential for mitigating chemical releases. If key chemical systems were located underground, where pathways to the surface are filtered, then the frequency of unfiltered chemical releases would be below the nominal release frequency.

To model the situation where the chemical facility is underground, we set the probability of structural damage from hydrogen detonation scenarios to zero. Further, we set the conditional probability of having a hydrogen explosion (which causes the chemical release) to zero. Under these two assumptions, the mean core damage risk decreases to 8E-11/yr, resulting in a negligible core damage frequency. The mean chemical dispersion release frequency decreased to 8E-4/yr.

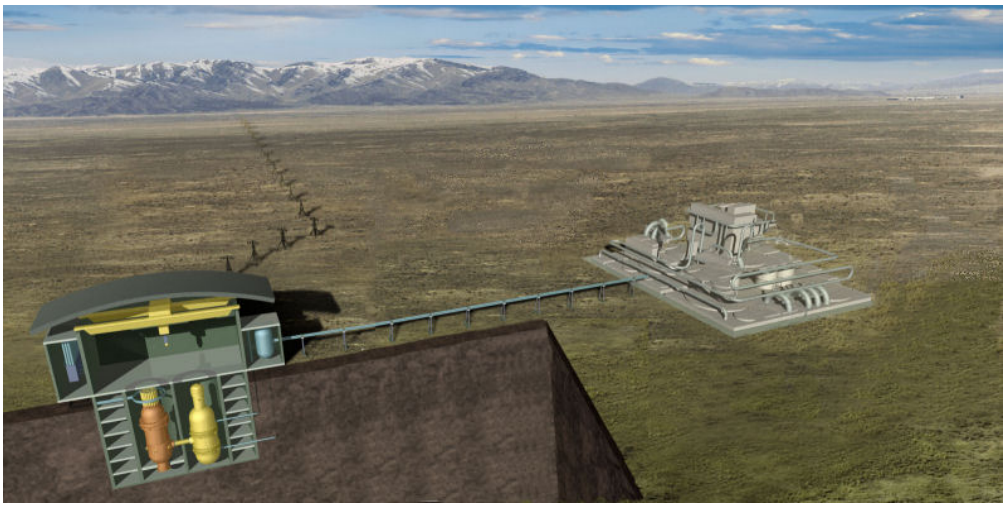

Figure 33. Case 5: Constructing the chemical facility primarily underground.

3.6 Case 6: Move the Control Room Offsite

As shown in Figure 34, in this sensitivity case we consider the possibility of moving the nuclear facility control room offsite, where *offsite* implies that the control room would be at a sufficient distance to preclude impacts from chemical dispersion events. This option does not affect the core damage risk from hydrogen events, but does mitigate the impacts from dispersion scenarios. By locating the control room portion of the plant away from the chemical facility at a significant distance (greater than 500 m), the evaluation and guidance provided by RG 1.78 would be met and would minimize control room habitability as one of the potential licensing issues. Further, if the HTGR design requires multiple reactors, a single centralized control room location shared by the multiple units may be able to use the segregation of the control room as an efficiency measure.

Figure 34. Case 6: Moving the nuclear plant control room offsite.

3.7 Summary of the Sensitivity Cases

The results of the sensitivity cases are shown in Table 7, where we list the mean core damage frequency and dispersion frequency for the cases. A couple of observations can be extracted from these cases. First, implementing a passive barrier (for example, earth or blast panels) between the chemical facility and the nuclear plant yields a significant decrease in the core damage frequency. Second, several of the cases did little to improve the dispersion frequency (or the associated consequences) of chemical releases, simply due to the difficulty in containing dispersion events. The one case that had the largest improvement relating to the dispersion risk was that of relocating the nuclear control room offsite. Third, when the separation distance is less than 100 m, the core damage frequency is large enough to result in possible licensing issues in a risk-informed environment such as that described be RG 1.174—hence, the need for either mitigating the risk by using one of the analyzed options or increasing the separation distance to at least 110 m.

Another item to consider in mitigating risk related to chemical dispersion events is to not use hydrogen-production chemical cycles that use hazardous chemicals. For example, the electrolysis option, while still producing hydrogen (and incurring the core damage risk associated with that gas), would eliminate the dispersion risk and reduce the need to relocate the reactor control room.

Table 7. Summary results from the sensitivity cases.

Case	Description	Mean Core Damage Frequency (per year)	Mean Chemical Dispersion Frequency (per year)
Nominal	No mitigating features and the separation distance set at 60 m	7E-6	3E-3
1.	Varying the separation distance between the nuclear and hydrogen production facilities	See Figure 29	3E-3
2.	Constructing an earthen barrier between the nuclear and chemical facilities	4E-10	3E-3
3.	Constructing the nuclear facility primarily underground	4E-10	3E-3
4.	Constructing blast panels near the chemical facility to dampen overpressure events	4E-10	3E-3
5.	Constructing the chemical facility primarily underground	8E-11	8E-4
6.	Moving the nuclear plant control room offsite	7E-6	Releases no longer impact control room

4. CONCLUSIONS

This report provides an overview of the objectives, methods, and models used to evaluate locating a hydrogen production facility near a nuclear power plant (e.g., very-high-temperature gas reactor). In order to answer safety related questions for this nuclear and chemical facility, we utilized standard PSA methodologies to answer three questions: (1) what can happen, (2) how likely is it, and (3) what are the consequences? As part of answering these questions, we developed a model suitable to determine separation distances for hydrogen process structures and the nuclear plant structures.

To recall our motivations for this work, we iterate our primary and secondary objectives:

1. The primary objective of the model-development and analysis is to answer key safety questions related to the placement of one (or more) hydrogen production plant in the vicinity of one (or more) high-temperature nuclear reactor. From a thermal-hydraulic standpoint we would like the two facilities to be quite close. However, safety and regulatory implications force the separation distance to be increased, perhaps substantially. Without answering the safety questions posed in this report, the likelihood for obtaining a permit to construct and build such as facility in the U.S. would be questionable.
2. The secondary objective of the model-development and analysis is to provide decision makers with tools and insights into potential engineering designs that may mitigate hazards inherent in the coupling of nuclear and hydrogen production facilities. These safety-based tools allow for the selection of a variety of facility types (e.g., different chemical processes, safety barriers) in addition to the ability to estimate risk as the separation distance is modified.

The quantitative analysis performed for this report provides us with a scoping mechanism to determine key design parameters relating to the development of a nuclear-based hydrogen production facility. From the quantitative analysis and the assumptions that went into the risk evaluation, we have determined recommendations that should be considered as the facility design is refined. These recommendations are based upon the risk analysis results, including potential changes to the nominal risk (see Table 8) as facility modification are made to mitigate risk. From our sensitivity calculations, we see that when the separation distance is less than 100 m, the core damage frequency is large enough (greater than 1E-6/yr) to become problematic in a risk-informed environment.

Table 8. Mean risk results for the hydrogen plant SAPHIRE risk model.

Case	Core Damage Frequency (per year)	Dispersion Frequency (per year)
Nominal (separation distance of 60 m)	7E-6	3E-3

By using a combination of risk-mitigating designs, for example placing an earth blast barrier between the chemical facility and the nuclear plant and moving the control room offsite, the separation distance can be decreased. However, as the separation distance is decreased, the risk-mitigating features may become harder to construct. In the case of the earth barrier, a limited horizontal distance would translate into a limited vertical distance, which may not be large enough to effectively prevent damage from hydrogen detonation events.

- We recommend that a separation distance of between 60 to 120 m be considered for the base design of the hydrogen production facility. This separation distance should be coupled with a blast barrier(s) sufficient to prevent hydrogen detonation events from damaging the nuclear structures. In lieu of any mitigating barriers, the separation distance should be at least 110 m.

- We recommend that as hydrogen is produced, it is transferred offsite for compression and storage, thereby limiting the overall on-site storage (i.e., near the nuclear facility) in a single location to less than 100 kg of hydrogen. In general, there should be a conscientious effort in the chemical facility design process to minimize the storage of combustible gasses near the point of production.
- We recommend the hydrogen distribution system use an inerted (i.e., low oxygen concentration), co-axial piping system to minimize the potential for hydrogen leaks resulting in either a deflagration or detonation. Hydrogen is a difficult gas to prevent from leaking, so a passive design should funnel leaking hydrogen away from critical areas of the plant.
- We recommend that the control room for the nuclear plant be located offsite and away from the chemical hazards posed by the hydrogen production facility. The offsite control room design should consider redundant data paths to the reactor to offset risks associated with locating the control room away from the nuclear plant.
- We recommend that a detailed PSA be performed for the VHTGR and the chemical facility as the design details become mature. For the analysis in this report, we had to rely on a limited PSA, performed over 20 years ago on a similar type of plant—this older plant type may have limited applicability to newer designs.
- We recommend that in addition to the major nuclear plant structures and systems being constructed below ground level, that where possible critical portions of the chemical facility be considered for belowground placement. The definition of *critical* in relation to the chemical facility refers to those parts of the process equipment that contain significant quantities of either hazardous chemicals or provide storage for combustible gas such as hydrogen.

By considering a variety of passive design measures aimed at mitigating risk, it is possible to ensure that adequate safety levels are maintained for the nuclear plant. Next-generation advanced reactors such as the VHTGR are being designed to be safer than the current fleet of light-water reactors over a wide variety of upset conditions, including those conditions that may be seen from the adjoining hydrogen production facility.

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Appendix A

Potential Chemical Hazards

Appendix A

Potential Chemical Hazards

A-1. Introduction

Generating hydrogen using a thermochemical process requires both heat and various chemicals, some of which might be hazardous. The concern partially addressed here is the hazard these various chemicals might pose to safety if an accident were to occur and a quantity of these chemicals were released into the atmosphere. The approach taken to address this concern comprises a series of steps:

1. Survey the different thermochemical processes that have been proposed for generating hydrogen
2. List the chemicals used in each process
3. Compare this list of chemicals to those identified by the U.S. Environmental Protection Agency (EPA) as Extremely Hazardous Substances (EHS)
4. Analyze the chemicals that appear on both lists, using the ALOHA code to assess the hazard from a release of one of these chemicals into the atmosphere.

A-2. Survey of Thermochemical Processes

General Atomics led a program funded by a DOE NERI grant, titled High Efficiency Generation of Hydrogen Fuels Using Nuclear Power [GA 2003], that included an evaluation of many proposed thermochemical processes. That program first identified and organized over 100 thermochemical water-splitting cycles, then performed a screening that reduced the number to 25, which were then the subjects of more detailed study. The 25 processes that survived the initial screening are listed in Table A-1.

Table A-1. Water-splitting thermochemical processes.

Cycle	Name	Reaction
1	Westinghouse	$2H_2SO_4(g) \rightarrow 2SO_2(g) + 2H_2O(g) + O_2(g)$ $SO_2(g) + 2H_2O(a) \rightarrow H_2SO_4(a) + H_2(g)$
2	Ispra Mark	$2H_2SO_4(g) \rightarrow 2SO_2(g) + 2H_2O(g) + O_2(g)$ $2HBr(a) \rightarrow Br_2(a) + H_2(g)$ $Br_2(l) + SO_2(g) + 2H_2O(l) \rightarrow 2HBr(g) + H_2SO_4(a)$
3	UT-3 Univ. of Tokyo	$2Br_2(g) + 2CaO \rightarrow 2CaBr_2 + O_2(g)$ $3FeBr_2 + 4H_2O \rightarrow Fe_3O_4 + 6HBr + H_2(g)$ $CaBr_2 + H_2O \rightarrow CaO + 2HBr$ $Fe_3O_4 + 8HBr \rightarrow Br_2 + 3FeBr_2 + 4H_2O$
4	Sulfur-Iodine	$2H_2SO_4(g) \rightarrow 2SO_2(g) + 2H_2O(g) + O_2(g)$ $2HI \rightarrow I_2(g) + H_2(g)$ $I_2 + SO_2(a) + 2H_2O \rightarrow 2HI(a) + H_2SO_4(a)$
5	Julich Center EOS	$2Fe_3O_4 + 6FeSO_4 \rightarrow 6Fe_2O_3 + 6SO_2 + O_2(g)$ $3FeO + H_2O \rightarrow Fe_3O_4 + H_2(g)$ $Fe_2O_3 + SO_2 \rightarrow FeO + FeSO_4$
6	Tokyo Inst. Tech. Ferrite	$2MnFe_2O_4 + 3Na_2CO_3 + H_2O \rightarrow 2Na_3MnFe_2O_6 + 3CO_2(g) + H_2(g)$ $4Na_3MnFe_2O_6 + 6CO_2(g) \rightarrow 4MnFe_2O_4 + 6Na_2CO_3 + O_2(g)$

Cycle	Name	Reaction
7	Hallett Air Products	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2HCl \rightarrow Cl_2(g) + H_2(g)$
8	Gaz de France	$2K + 2KOH \rightarrow 2K_2O + H_2(g)$
		$2K_2O \rightarrow 2K + K_2O_2$
		$2K_2O_2 + 2H_2O \rightarrow 4KOH + O_2(g)$
9	Nickel Ferrite	$NiMnFe_4O_6 + 2H_2O \rightarrow NiMnFe_4O_8 + 2H_2(g)$
		$NiMnFe_4O_8 \rightarrow NiMnFe_4O_6 + O_2(g)$
10	Aachen Univ. Julich	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2CrCl_2 + 2HCl \rightarrow 2CrCl_3 + H_2(g)$
		$2CrCl_3 + 2CrCl_2 + Cl_2(g)$
11	Ispra Mark 1C	$2CuBr_2 + Ca(OH)_2 \rightarrow 2CuO + 2CaBr_2 + H_2O$
		$4CuO(s) \rightarrow 2Cu_2O(s) + O_2(g)$
		$CaBr_2 + 2H_2O \rightarrow Ca(OH)_2 + 2HBr$
		$Cu_2O + 4HBr \rightarrow 2CuBr_2 + H_2(g) + H_2O$
12	LASL-U	$3CO_2 + U_3O_8 + H_2O \rightarrow 3UO_2CO_3 + H_2(g)$
		$3UO_2CO_3 \rightarrow 3CO_2(g) + 3UO_3$
		$6UO_3(s) \rightarrow 2U_3O_8(s) + O_2(g)$
13	Ispra Mark 8	$3MnCl_2 + 4H_2O \rightarrow Mn_3O_4 + 6HCl + H_2(g)$
		$3MnO_2 \rightarrow Mn_3O_4 + O_2(g)$
		$4HCl + Mn_3O_4 \rightarrow 2MnCl_2(a) + MnO_2 + 2H_2O$
14	Ispra Mark 6	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2CrCl_2 + 2HCl \rightarrow 2CrCl_3 + H_2(g)$
		$2CrCl_3 + 2FeCl_2 \rightarrow 2CrCl_2 + 2FeCl_3$
		$2FeCl_3 \rightarrow Cl_2(g) + 2FeCl_2$
15	Ispra Mark 4	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2FeCl_2 + 2HCl + S \rightarrow 2FeCl_3 + H_2S$
		$2FeCl_3 \rightarrow Cl_2(g) + 2FeCl_2$
		$H_2S \rightarrow S + H_2(g)$
16	Ispra Mark 3	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2VOCl_2 + 2HCl \rightarrow 2VOCl_3 + H_2(g)$
		$2VOCl_3 \rightarrow Cl_2(g) + 2VOCl_2$
17	Ispra Mark 2	$Na_2O.MnO_2 + H_2O \rightarrow 2NaOH(a) + MnO_2$
		$4MnO_2(s) \rightarrow 2Mn_2O_3(s) + O_2(g)$
		$Mn_2O_3 + 4NaOH \rightarrow 2Na_2O.MnO_2 + H_2(g) + H_2O$
18	Ispra CO/Mn_3O_4	$6Mn_2O_3 \rightarrow 4Mn_3O_4 + O_2(g)$
		$C(s) + H_2O(g) \rightarrow CO(g) + H_2(g)$
		$CO(g) + 2Mn_3O_4 \rightarrow C + 3Mn_2O_3$
19	Ispra Mark 7B	$2Fe_2O_3 + 6Cl_2(g) \rightarrow 4FeCl_3 + 3O_2(g)$
		$2FeCl_3 \rightarrow Cl_2(g) + 2FeCl_2$
		$3FeCl_2 + 4H_2O \rightarrow Fe_3O_4 + 6HCl + H_2(g)$
		$4Fe_3O_4 + O_2(g) \rightarrow 6Fe_2O_3$
		$4HCl + O_2(g) \rightarrow 2Cl_2(g) + 2H_2O$

Cycle	Name	Reaction
20	Vanadium Chloride	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2HCl + 2VCl_2 \rightarrow 2VCl_3 + H_2(g)$
		$2VCl_3 \rightarrow VCl_4 + VCl_2$
		$2VCl_4 \rightarrow Cl_2(g) + 2VCl_3$
21	Mark 7A	$2FeCl_3(l) \rightarrow Cl_2(g) + 2FeCl_2$
		$3FeCl_2 + 4H_2O(g) \rightarrow Fe_3O_4 + 6HCl(g) + H_2(g)$
		$4Fe_3O_4 + O_2(g) \rightarrow 6Fe_2O_3$
		$6Cl_2(g) + 2Fe_2O_3 \rightarrow 4FeCl_3(g) + 3O_2(g)$
		$Fe_2O_3 + 6HCl(a) \rightarrow 2FeCl_3(a) + 3H_2O(l)$
22	GA Cycle 23	$H_2S(g) \rightarrow S(g) + H_2(g)$
		$2H_2SO_4(g) \rightarrow 2SO_2(g) + 2H_2O(g) + O_2(g)$
		$3S + 2H_2O(g) \rightarrow 2H_2S(g) + SO_2(g)$
		$3SO_2(g) + 2H_2O(l) \rightarrow 2H_2SO_4(a) + S$
		$S(g) + O_2(g) \rightarrow SO_2(g)$
23	US-Chlorine	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2CuCl + 2HCl \rightarrow 2CuCl_2 + H_2(g)$
		$2CuCl_2 \rightarrow 2CuCl + Cl_2(g)$
24	Ispra Mark 9	$2FeCl_3 \rightarrow Cl_2(g) + 2FeCl_2$
		$3Cl_2(g) + 2Fe_3O_4 + 12HCl \rightarrow 6FeCl_3 + 6H_2O + O_2(g)$
		$3FeCl_2 + 4H_2O \rightarrow Fe_3O_4 + 6HCl + H_2(g)$
25	Ispra Mark 6C	$2Cl_2(g) + 2H_2O(g) \rightarrow 4HCl(g) + O_2(g)$
		$2CrCl_2 + 2HCl \rightarrow 2CrCl_3 + H_2(g)$
		$2CrCl_3 + 2FeCl_2 \rightarrow 2CrCl_2 + 2FeCl_3$
		$2CuCl_2 \rightarrow 2CuCl + Cl_2(g)$
		$CuCl + FeCl_3 \rightarrow CuCl_2 + FeCl_2$

A-3. Chemical List

Each of the 25 water-splitting thermochemical processes was reviewed, and the constituent chemicals used in the processes were identified. This review produced a total of 58 unique chemicals. This list of chemicals is presented in Table A-2.

Table A-2. Chemicals found in water-splitting thermochemical processes.

CAS[a] Number	Formula	Chemical Name	Molecular Weight (g/mole)
10025-73-7	Cr-Cl3	CHROMIC CHLORIDE	158.35
10034-85-2	H-I	hydrogen iodide	127.912
10035-10-6	H-Br	Hydrogen bromide	80.912
10049-05-5	Cr-Cl2	CHROMOUS CHLORIDE	122.9
10213-09-9	V-O-Cl2	vanadyl dichloride	137.85
10580-52-6	V-Cl2	vanadium dichloride	121.85
12136-45-7	K2-O	POTASSIUM MONOXIDE	94.1794
124-38-9	C-O2	carbon dioxide	44.01
1305-62-0	Ca-(OH)2	Calcium hydroxide	74.09
1305-78-8	Ca-O	Calcium oxide	56.0794
1309-37-1	Fe2-O3	Iron(III) oxide	159.69

CAS[a] Number	Formula	Chemical Name	Molecular Weight (g/mole)
1310-58-3	K-O-H	POTASSIUM HYDROXIDE	56.106
1310-73-2	Na-O-H	sodium hydroxide	39.997
1313-13-9	Mn-O2	Manganese(IV) oxide	86.94
1317-34-6	Mn2-O3	Manganese(III) oxide	157.8742
1317-35-7	Mn3-O4	Manganese Tetroxide (AS MN)	228.81
1317-38-0	Cu-O	Cupric oxide	79.5454
1317-39-1	Cu2-O	Cuprous oxide	143.0914
1317-61-9	Fe3-O4	Iron(II, III) oxide	231.5326
1333-74-0	H2	Hydrogen	2.016
1344-58-7	U-O3	Uranium trioxide	286.03
1344-59-8	U3-O8	Uranium(V,VI) oxide (pitchblende)	842.08
1345-25-1	Fe-O	Iron(II) oxide	71.8444
17014-71-0	K2-O2	POTASSIUM PEROXIDE	110.1788
497-19-8	Na2-C-O3	Sodium carbonate	105.99
630-08-0	C-O	carbon monoxide	28.01
7440-09-7	K	potassium	39.098
7440-44-0	C	carbon	12.011
7446-09-5	S-O2	SULFUR DIOXIDE	64.065
7447-39-4	Cu-Cl2	cupric chloride	134.451
7553-56-2	I2	iodine	253.809
7632-51-1	V-Cl4	vanadium tetrachloride	192.752
7647-01-0	H-Cl	hydrogen chloride	36.461
7664-93-9	H2-S-O4	SULFURIC ACID	98.079
7446-11-9	S-O3	Sulfur trioxide	80.06
7704-34-9	S	sulfur	32.066
7705-08-0	Fe-Cl3	FERRIC CHLORIDE	162.205
7718-98-1	V-Cl3	vanadium trichloride	157.3
7720-78-7	Fe-S-O4	Ferrous sulfate	151.9026
7726-95-6	Br2	Bromine	159.808
7727-18-6	V-O-Cl3	vanadium oxytrichloride	173.299
7732-18-5	H2-O	Water	18.015
7758-89-6	Cu-Cl	cuprous chloride	98.999
7758-94-3	Fe-Cl2	FERROUS CHLORIDE	126.752
7773-01-5	Mn-Cl2	manganese chloride	125.843
7782-44-7	O2	Oxygen	31.999
7782-50-5	Cl2	CHLORINE	70.905
7783-06-4	H2-S	hydrogen sulfide	34.082
7789-41-5	Ca-Br2	Calcium Bromide	199.89
7789-45-9	Cu-Br2	COPPER BROMIDE	223.354
7789-46-0	Fe-Br2	Ferrous bromide	215.653
--[b]	Mn-Fe2-O4	--	
--	Na2-O.-Mn-O2	--	
--	Na3-Mn-Fe2-O6	--	
--	Ni-Mn-Fe4-O6	--	
--	Ni-Mn-Fe4-O8	--	
--	U-O2-C-O3	uranyl carbonate (rutherfordine)	330.08

a. Chemical Abstract Service.
b. No information on this chemical could be found.

A-4. Extremely Hazardous Substances

The list of chemicals from Table A-2 was then compared to the extremely hazardous substances (EHS) list [40 CFR 355] maintained by the U. S. Environmental Protection Agency (EPA). The EPA list of EHS identifies those chemicals whose presence is required to be reported to local officials for emergency planning purposes if the quantity exceeds a certain threshold. This quantity is defined as the threshold planning quantity (TPQ). The EPA also provides guidance on concentrations of EHSs that are of concern. Over the last 20 years or so, various limits have been proposed from a number of different organizations. However, the recent practice being embraced by both EPA and DOE is to adopt the Emergency Response Planning Guidelines (ERPGs) developed by American Industrial Hygiene Association (AIHA) (AIHA 2003). These ERPGs are defined in Table A-3.

Table A-3. Emergency Response Planning Guidelines (ERPG) definitions.

Level of Concern	Description
ERPG-1	Maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
ERPG-2	The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individuals ability to take protective action.
ERPG-3	The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

Table A-4 lists those chemicals from Table A-2 that are categorized by the EPA as EHS. Also listed on Table A-4 are the recommended concentrations for emergency response planning.

Each of the chemicals listed in Table A-4 were examined as a possible subject for atmospheric dispersion, given an accidental release into the environment. The ALOHA code [EPA 2004] was chosen as the tool for assessing the hazard from an atmospheric release of the chemicals listed in Table A-4. ALOHA was developed jointly by the National Oceanic and Atmospheric Administration (NOAA) and the U.S. EPA, and is designed especially for use by people responding to chemical accidents, as well as for emergency planning and training. Because it is an atmospheric dispersion analysis code, ALOHA is not for use with hazardous solid or liquid material. Two chemicals on Table A-4 fall into this category. Chromic chloride is a solid and sulfuric acid is a liquid. Therefore, ALOHA cannot be used to assess the hazard associated with these chemicals and were excluded from further analysis. Note that ALOHA does not model liquid vapor releases—this limitation will restrict the modeling of upsets where high-temperature sulfuric acid is released. However, we did look at dispersion of sulfur trioxide, which is released from high-temperature sulfuric acid.

Without site-specific or process-specific information on facility to be analyzed, worst-case conservative assumptions were made in performing the dispersion calculations. Guidance for these assumptions was taken from a hazard analysis guidance document developed by the EPA (EPA 1987). The guidance on worst-case analyses provided the inputs for such details as the characteristics of the release, the local climatic conditions and the surrounding terrain. However, the quantity of the EHS released still needed to be determined. For this input a simple assumption was made concerning the

Table A-4. Extremely hazardous substances (EHS) identified in Table A-2.

Formula	Chemical Name	TPQ (lbs)	Molecular Weight (g/mole)	ERPG-1 (ppm)	ERPG-1 (mg/m3)	ERPG-2 (ppm)	ERPG-2 (mg/m3)	ERPG-3 (ppm)	ERPG-3 (mg/m3)
Cr-Cl3	Chromic Chloride	1/10,000[b]	158.35	619	4	1160	7.5	11604	75
S-O2	Sulfur Dioxide	500	64.065	0.3	0.00078	3	0.00784	15	0.03922
H-Cl	Hydrogen Chloride	500	36.461	3	0.00446	20	0.02976	150	0.22323
H2-S-O4	Sulfuric Acid	1000	98.079	500	2	2498	10	7494	30
S-O3	Sulfur Trioxide[a]	100	80.06	612	2	3060	10	9181	30
Br2	Bromine	500	159.808	0.1	0.00065	0.5	0.00326	5	0.03261
Cl2	Chlorine	100	70.905	1	0.00289	3	0.00868	20	0.05788
H2-S	Hydrogen Sulfide	500	34.082	0.1	0.00014	30	0.04173	100	0.13911

a. Sulfur trioxide is not one of the chemicals listed on Table A-1. However, it is produced when sulfuric acid is heated (which it will be in a thermochemical hydrogen production process).

b. Lower quantity (1 pound) applies to solids in powder form with particulate size less than 100 microns, or if the substance is in solution or molten form. Otherwise the 10,000 pound TPQ limit applies.

Table A-5. ERPG-3 footprint (in km) for different levels of daily hydrogen production (in kg).

Cas Number	Formula	Chemical	Molecular Weight	EDBPI (kg)	Physical State	Temp		ERPG-3 Footprint (km) 25[a]	50[a]	100[a]	500[a]	1000[a]
10025-73-7	Cr-Cl3	Chromic Chloride	158.35	78,391	solid	1152	C (melting)	--[b]	--	--	--	--
7446-09-5	S-O2	Sulfur Dioxide	64.065	31,715	gas			2.1	2.7	3.6	7.1	9.7
7647-01-0	H-Cl	Hydrogen Chloride	36.461	18,050	gas			0.98	1.3	1.6	3.1	4.1
7664-93-9	H2-S-O4	Sulfuric Acid	98.079	48,554	liquid	290	C (boiling)	--[b]	--	--	--	--
7446-11-9	S-O3	Sulfur Trioxide	80.06	39,634	solid	45	C (boiling)	2.4	3.2	4.3	8.5	>10
7726-95-6	Br2	Bromine	159.808	79,113	liquid	59.5	C (boiling)	2.9	3.9	5.2	9.4	>10
7782-50-5	Cl2	Chlorine	70.905	35,101	gas			1.8	2.4	3.2	6.4	8.7
7783-06-4	H2-S	Hydrogen Sulfide	34.082	16,872	gas			1.2	1.5	2	3.7	4.9

a. Assumed daily production (in kg) of hydrogen (used as the basis for the quantity of EHS released into the atmosphere).

b. No calculations made since ALOHA does not treat solids or non-volatile liquids.

hydrogen generation process. It was assumed that the amount of EHS that could be release is equivalent to the amount of the chemical needed to produce a specified quantity of hydrogen in a daily batch process. In the following presentation of results, this quantity is referred to as the Equivalent Daily Batch Process Inventory (EDBPI). The atmospheric dispersion calculations were then performed assuming various quantities of daily hydrogen generation.

The final assumption relates to the concentration of the EHS that poses the level of concern (LOC) that is most relevant to the situation under consideration. Here, the LOC chosen is the EPRG-3 levels under the assumption that the population at risk is the work force of the nuclear plant providing heat for the thermochemical process.

Calculating the ERPG-3 levels for the EHS in the EDBPI quantities associated with a range of hydrogen production levels assuming worst-case release conditions yields the result presented in Table A-5 and in Figure A-1.

A sensitivity calculation was performed to estimate the effect of varying the input values from worst-case to a more realistic situation. Using bromine and worst-case weather and release characteristics, produces the ERPG footprint shown in Figure A-2. Figure A-3 illustrates the result of changing from a worst-case calculation to more realistic release and weather pattern. In this sensitivity case, bromine is used assuming a quantity equivalent to 100 kg of hydrogen production. In the worst case calculation, the entire quantity of bromine (7,911 kg) is assumed to be instantaneously released into a stable slow wind (1.5 m/sec). The sensitivity case assumes the bromine is stored in a spherical tank, 2-meters in diameter, which has 100-cm diameter hole 25% from the bottom of the tank. Weather conditions are assumed to be more typical (slightly stronger wind and medium stability). Under these conditions, the ERPG footprint is reduced from the worst case ERPG-3 footprint of 5.2 km (see Table A-5 and Figure A-2) to 1.9 km. A second sensitivity case was then calculated that only changed the air temperature from 70^{o}F to 30^{o}F. This single change resulted in a further reduction of the ERPG footprint (shown on Figure A-4) from 1.9 km to 1.1 km.

A-5. Conclusions and Observations

Any significant release of a hazardous material will pose significant problems. However, not all thermochemical processes used for generating hydrogen use hazardous materials. Further, the quantity of material stored on-site to support the hydrogen generation process is still an unknown. Simplistic assumptions were made to derive the quantities used here. For our analysis, the quantity assumed is based on the amount needed for to produce a specified quantity of hydrogen in one day in a batch type of process. Also, since the total quantity of material is not likely to be stored in a single location (as assumed here), further sensitivity studies will help to assess more realistic situations. Nevertheless, it appears prudent that wherever possible, any hazardous material should be stored in a down-wind location from the rest of the plant or the nuclear generating facility. Other passive measures of mitigating risks will be evaluated in a later report.

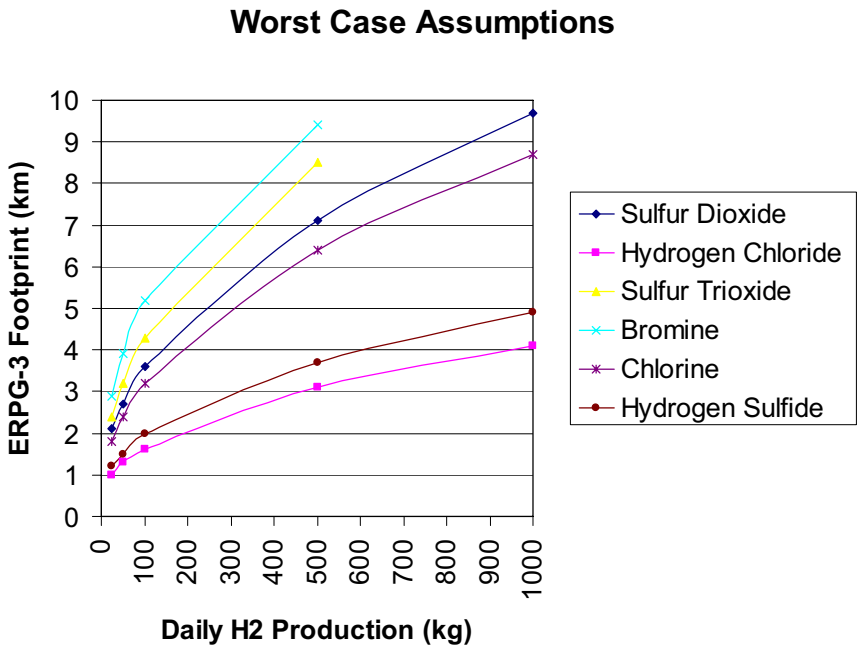

Figure A-1. Plot of ERPG-3 footprints (in km) for the EHS quantities associated with different levels of daily hydrogen production (in kg), assuming worst-case release conditions.

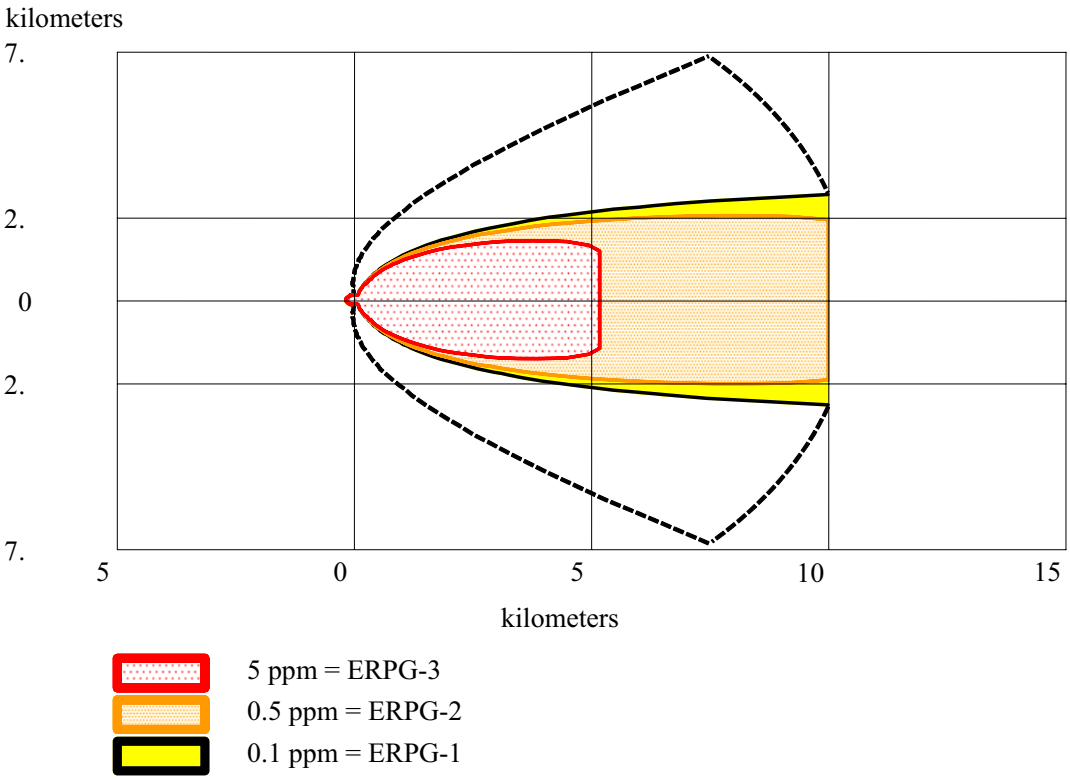

Figure A-2. ERPG footprint for bromine (7,911 kg, equivalent to 100 kg of hydrogen production) is shown, assuming worst case conditions. ERPG-3 distance is 5.2 km.

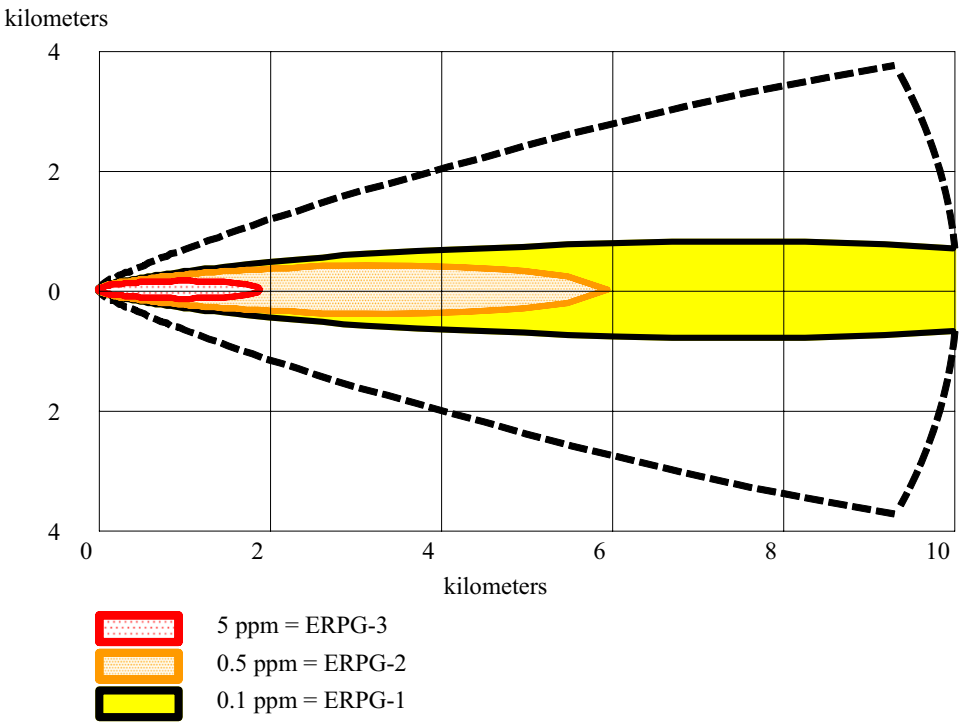

Figure A-3. ERPG footprint for bromine (7,911 kg, equivalent to 100 kg of hydrogen production), using realistic weather characteristics and bromine released from 2-m-diameter tank with 100-cm hole 25% from bottom of the tank.

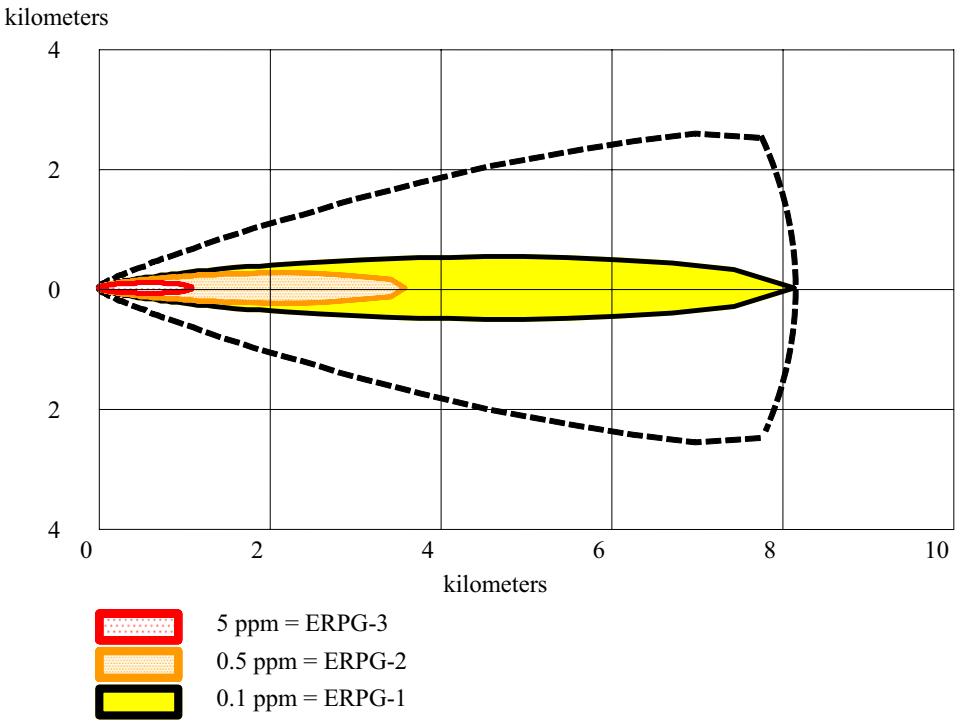

Figure A-4. Bromine ERPG footprint illustrating the effect of changing air temperature from 70ºF to 30ºF (compare to Figure A-2). ERPG-3 footprint reduces from 1.9 km to 1.1 km.

A-6. References

40 CFR 355, "Emergency Planning and Notification," *Code of Federal Regulations,* Office of the Federal Register, Revision Date November 26, 2004. (Appendices A and B list EHS alphabetically and by CAS number, respectively.)

AIHA 2003, *The AIHA 2003 Emergency Response Planning Guidelines and Workplace Environmental Exposure Level Guides Handbook*, AIHA Press, Fairfax, Virgina, 2003.

EPA 1987, *Technical Guidance for Hazard Analysis – Emergency Planning for Extremely Hazardous Substances*, U.S. Environmental Protection Agency, Federal Emergency Management Agency, U.S. Department of Transportation, December 1987.

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GA 2003, L. C. Brown, et al, High Efficiency Generation of Hydrogen Fuels Using Nuclear Power – Final Technical Report for the Period August 1, 1999 through September 30, 2002, General Atomics, GA-A24285, Nuclear Energy Research Initiative (NERI) Program Grant No. DE-FG03-99SF21888, June 2003.

Appendix B

Basic Event Parameters from the SAPHIRE Hydrogen Facility Risk Model

Appendix B

Basic Event Parameters from the SAPHIRE Hydrogen Facility Risk Model

In this appendix, we provide information on the SAPHIRE risk model basic events, including the event name, probability (or frequency, per year, if it is an initiating event), uncertainty distribution type and parameters, description, 95th percentile value of the uncertainty distribution, and supplementary notes, as illustrated below.

NAME Description *Note:*	Probability	Uncertainty 95th percentile
CD-PLTUPSET CORE DAMAGE GIVEN A PLANT UPSET *Note:*	1.00E-07	Constrained non-informative 95th = 4.35E-7
CORE-DAMGAGE-1 PROB OF CD GIVEN STRUCT FAILURE *Note:*	2.00E-02	Constrained non-informative 95th = 8.65E-2
CS-STRUCT-DAM NPP STRUCTURAL DAMAGE *Note:*	9.75E-02	Constrained non-informative 95th = 4.04E-1
EXPL-H2LEAK EXPLOSION DUE TO HYDROGEN LEAK *Note:*	2.50E-01	Beta(6.5, 19.5) 95th = 3.98E-1
GLEVEL1 SEISMIC G-LEVEL 1 *Note:*	8.33E-04	Constrained non-informative 95th = 3.62E-3
GLEVEL2 SEISMIC G-LEVEL 2 *Note:*	9.91E-05	Constrained non-informative 95th = 4.31E-4
GLEVEL3 SEISMIC G-LEVEL 3 *Note:*	2.41E-05	Constrained non-informative 95th = 1.05E-4
H2S-ACC-EQ-1S1 ACCUMULATOR 1 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-ACC-EQ-SEIS1 template event.	1.45E-12	Constrained non-informative 95th = 6.30E-12
H2S-ACC-EQ-1S2 ACCUMULATOR 1 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-ACC-EQ-SEIS2 template event.	1.11E-06	Constrained non-informative 95th = 4.84E-6
H2S-ACC-EQ-1S3 ACCUMULATOR 1 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-ACC-EQ-SEIS3 template event.	1.43E-03	Constrained non-informative 95th = 6.22E-3
H2S-ACC-EQ-2S1	1.45E-12	Constrained non-informative

ACCUMULATOR 2 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-ACC-EQ-SEIS1 template event.		95th = 6.30E-12
H2S-ACC-EQ-2S2 ACCUMULATOR 2 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-ACC-EQ-SEIS2 template event.	**1.11E-06**	Constrained non-informative 95th = 4.84E-6
H2S-ACC-EQ-2S3 ACCUMULATOR 2 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-ACC-EQ-SEIS3 template event.	**1.43E-03**	Constrained non-informative 95th = 6.22E-3
H2S-ACC-OP-1 H2 ACCUMULATOR 1 FAILS DUE TO OVERPRESSURIZATION *Note:*	**5.00E-09**	Lognormal(EF=30.0) 95th = 1.77E-8
H2S-ACC-OP-2 H2 ACCUMULATOR 2 FAILS DUE TO OVERPRESSURIZATION *Note:*	**5.00E-09**	Lognormal(EF=30.0) 95th = 1.77E-8
H2S-CHL-ACCDIS COOLING UNIT RUPTURE *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-CHL-COMP1 COMPRESSOR 1 COOLING UNIT RUPTURE *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-CHL-COMP2 COMPRESSOR 2 COOLING UNIT RUPTURE *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-CHL-COMP3 COMPRESSOR 3 COOLING UNIT RUPTURE *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-CHL-EQ-ADS1 ACC COOLING UNIT SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS1 template event.	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
H2S-CHL-EQ-ADS2 ACC COOLING UNIT SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS2 template event.	**1.81E-05**	Constrained non-informative 95th = 7.87E-5
H2S-CHL-EQ-ADS3 ACC COOLING UNIT SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS3 template event.	**8.47E-03**	Constrained non-informative 95th = 3.67E-2
H2S-CHL-EQ-COMP1S1 COMP 1 COOLING UNIT SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS1 template event.	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
H2S-CHL-EQ-COMP1S2 COMP 1 COOLING UNIT SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS2 template event.	**1.81E-05**	Constrained non-informative 95th = 7.87E-5
H2S-CHL-EQ-COMP1S3 COMP 1 COOLING UNIT SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS3 template event.	**8.47E-03**	Constrained non-informative 95th = 3.67E-2
H2S-CHL-EQ-COMP2S1 COMP 2 COOLING UNIT SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS1 template event.	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
H2S-CHL-EQ-COMP2S2	**1.81E-05**	Constrained non-informative

COMP 2 COOLING UNIT SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS2 template event.		95th = 7.87E-5
H2S-CHL-EQ-COMP2S3 COMP 2 COOLING UNIT SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS3 template event.	**8.47E-03**	Constrained non-informative 95th = 3.67E-2
H2S-CHL-EQ-COMP3S1 COMP 3 COOLING UNIT SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS1 template event.	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
H2S-CHL-EQ-COMP3S2 COMP 3 COOLING UNIT SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS2 template event.	**1.81E-05**	Constrained non-informative 95th = 7.87E-5
H2S-CHL-EQ-COMP3S3 COMP 3 COOLING UNIT SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS3 template event.	**8.47E-03**	Constrained non-informative 95th = 3.67E-2
H2S-MDC-RP-COMP1 COMPRESSOR 1 LEAKS *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-MDC-RP-COMP2 COMPRESSOR 2 LEAKS *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-MDC-RP-COMP3 COMPRESSOR 3 LEAKS *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
H2S-MOV-EQ-A10S1 ACC MOV 1 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-MOV-EQ-A10S2 ACC MOV 1 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
H2S-MOV-EQ-A10S3 ACC MOV 1 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
H2S-MOV-EQ-A20S1 ACC MOV 2 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-MOV-EQ-A20S2 ACC MOV 2 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
H2S-MOV-EQ-A20S3 ACC MOV 2 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
H2S-MOV-EQ-V1IS1 PV INLET MOV 1 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-MOV-EQ-V1IS2 PV INLET MOV 1 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
H2S-MOV-EQ-V1IS3 PV INLET MOV 1 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3

H2S-MOV-EQ-V1OS1 PV OUTLET MOV 1 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-MOV-EQ-V1OS2 PV OUTLET MOV 1 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
H2S-MOV-EQ-V1OS3 PV OUTLET MOV 1 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
H2S-MOV-EQ-V2IS1 PV INLET MOV 2 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-MOV-EQ-V2IS2 PV INLET MOV 2 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
H2S-MOV-EQ-V2IS3 PV INLET MOV 2 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
H2S-MOV-EQ-V2OS1 PV OUTLET MOV 2 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-MOV-EQ-V2OS2 PV OUTLET MOV 2 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
H2S-MOV-EQ-V2OS3 PV OUTLET MOV 2 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
H2S-MOV-RP-1I PRESSURE VESSEL INLET MOV 1 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-MOV-RP-1O PRESSURE VESSEL OUTLET MOV 1 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-MOV-RP-2I PRESSURE VESSEL INLET MOV 2 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-MOV-RP-2O PRESSURE VESSEL OUTLET MOV 2 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-MOV-RP-A1O ACCUMULATOR 1 OUTLET MOV 1 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-MOV-RP-A2O ACCUMULATOR 2 OUTLET MOV 1 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-PIP-EQ-H2S HYDROGEN SYS PIPING SIESMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-PIP-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
H2S-PIP-EQ-H2S2 HYDROGEN SYS PIPING SIESMIC FRAGILITY (0.3g)	**1.04E-05**	Constrained non-informative 95th = 4.52E-5

Note: This event uses the TEMPEV-PIP-EQ-SEIS2 template event.

H2S-PIP-EQ-H2S3 HYDROGEN SYS PIPING SIESMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-PIP-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
H2S-PIP-RP-HYDRO HYDROGEN SYSTEM PIPING RUPTURE *Note:*	**3.00E-08**	Lognormal(EF=10.0) 95th = 1.13E-7
H2S-TNK-EQ-1S1 PRESS TANK 1 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-VES-EQ-SEIS1 template event.	**6.06E-11**	Constrained non-informative 95th = 2.64E-10
H2S-TNK-EQ-1S2 PRESS TANK 1 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-VES-EQ-SEIS2 template event.	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
H2S-TNK-EQ-1S3 PRESS TANK 1 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-VES-EQ-SEIS3 template event.	**2.98E-03**	Constrained non-informative 95th = 1.29E-2
H2S-TNK-EQ-2S1 PRESS TANK 2 SEISMIC FRAGILITY (0.1g) *Note:* This event uses the TEMPEV-VES-EQ-SEIS1 template event.	**6.06E-11**	Constrained non-informative 95th = 2.64E-10
H2S-TNK-EQ-2S2 PRESS TANK 2 SEISMIC FRAGILITY (0.3g) *Note:* This event uses the TEMPEV-VES-EQ-SEIS2 template event.	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
H2S-TNK-EQ-2S3 PRESS TANK 2 SEISMIC FRAGILITY (0.6g) *Note:* This event uses the TEMPEV-VES-EQ-SEIS3 template event.	**2.98E-03**	Constrained non-informative 95th = 1.29E-2
H2S-TNK-RP-1 PRESSURE VESSEL 1 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
H2S-TNK-RP-2 PRESSURE VESSEL 2 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
HUMAN_ERROR Frequency of Hydrogen Events from Human Error *Note:*	**1.08E-02**	Gamma(5.5, .002) 95th = 1.93E-2
SFA-BUR-EQ-SHS1 GAS BURNER SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-VES-EQ-SEIS1 template event.	**6.06E-11**	Constrained non-informative 95th = 2.64E-10
SFA-BUR-EQ-SHS2 GAS BURNER SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-VES-EQ-SEIS2 template event.	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
SFA-BUR-EQ-SHS3 GAS BURNER SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-VES-EQ-SEIS3 template event.	**2.98E-03**	Constrained non-informative 95th = 1.29E-2
SFA-BUR-RP-SHEL GAS SYSTEM BURNER SHELL FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
SFA-CWS-EQ-TWS1 COOLING TOWER SEISMIC FRAGILITY (0.1) *Note:*	**0.00E+00**	No distribution assigned to this event.
SFA-CWS-EQ-TWS2	**0.00E+00**	No distribution

COOLING TOWER SEISMIC FRAGILITY (0.3) *Note:*		assigned to this event.
SFA-CWS-EQ-TWS3 COOLING TOWER SEISMIC FRAGILITY (0.6) *Note:*	**0.00E+00**	No distribution assigned to this event.
SFA-CWS-FC-TWR COOLING TOWER FAILS *Note:*	**0.00E+00**	Lognormal(EF=30.0) 95th = 0.00E+0
SFA-FAN-EQ-F1S1 AIR BLOWER 1 SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-FAN-EQ-SEIS1 template event.	**6.06E-11**	Constrained non-informative 95th = 2.64E-10
SFA-FAN-EQ-F1S2 AIR BLOWER 1 SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-FAN-EQ-SEIS2 template event.	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
SFA-FAN-EQ-F1S3 AIR BLOWER 1 SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-FAN-EQ-SEIS3 template event.	**2.98E-03**	Constrained non-informative 95th = 1.29E-2
SFA-FAN-EQ-F2S1 AIR BLOWER 2 SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-FAN-EQ-SEIS1 template event.	**6.06E-11**	Constrained non-informative 95th = 2.64E-10
SFA-FAN-EQ-F2S2 AIR BLOWER 2 SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-FAN-EQ-SEIS2 template event.	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
SFA-FAN-EQ-F2S3 AIR BLOWER 2 SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-FAN-EQ-SEIS3 template event.	**2.98E-03**	Constrained non-informative 95th = 1.29E-2
SFA-FAN-FC-1 AIR BLOWER FAN 1 FAILS *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
SFA-FAN-FC-2 AIR BLOWER FAN 2 FAILS *Note:*	**3.00E-07**	Lognormal(EF=10.0) 95th = 1.13E-6
SFA-HTX-EQ-AC1S1 ACID HTX 1 SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS1 template event.	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
SFA-HTX-EQ-AC1S2 ACID HTX 1 SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS2 template event.	**1.81E-05**	Constrained non-informative 95th = 7.87E-5
SFA-HTX-EQ-AC1S3 ACID HTX 1 SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS3 template event.	**8.47E-03**	Constrained non-informative 95th = 3.67E-2
SFA-HTX-EQ-AC21S1 ACID HTX 2 SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS1 template event.	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
SFA-HTX-EQ-AC21S2 ACID HTX 2 SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS2 template event.	**1.81E-05**	Constrained non-informative 95th = 7.87E-5
SFA-HTX-EQ-AC21S3 ACID HTX 2 SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-HTX-EQ-SEIS3 template event.	**8.47E-03**	Constrained non-informative 95th = 3.67E-2

SFA-HTX-FC-1 ACID COOLING HEAT EXCHANGER 1 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
SFA-HTX-FC-2 ACID COOLING HEAT EXCHANGER 2 FAILS *Note:*	**1.00E-07**	Lognormal(EF=10.0) 95th = 3.75E-7
SFA-MDP-EQ-AP1S1 ACID PUMP 1 SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-MDP-EQ-SEIS1 template event.	**4.05E-11**	Constrained non-informative 95th = 1.76E-10
SFA-MDP-EQ-AP1S2 ACID PUMP 1 SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-MDP-EQ-SEIS2 template event.	**1.11E-05**	Constrained non-informative 95th = 4.83E-5
SFA-MDP-EQ-AP1S3 ACID PUMP 1 SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-MDP-EQ-SEIS3 template event.	**6.23E-03**	Constrained non-informative 95th = 2.70E-2
SFA-MDP-EQ-AP2S1 ACID PUMP 2 SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-MDP-EQ-SEIS1 template event.	**4.05E-11**	Constrained non-informative 95th = 1.76E-10
SFA-MDP-EQ-AP2S2 ACID PUMP 2 SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-MDP-EQ-SEIS2 template event.	**1.11E-05**	Constrained non-informative 95th = 4.83E-5
SFA-MDP-EQ-AP2S3 ACID PUMP 2 SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-MDP-EQ-SEIS3 template event.	**6.23E-03**	Constrained non-informative 95th = 2.70E-2
SFA-MDP-RP-1 SULFURIC ACID PUMP 1 FAILS *Note:*	**1.00E-06**	Lognormal(EF=10.0) 95th = 3.75E-6
SFA-MDP-RP-2 SULFURIC ACID PUMP 2 FAILS *Note:*	**1.00E-06**	Lognormal(EF=10.0) 95th = 3.75E-6
SFA-MOV-EQ-AP1S1 PUMP 1 MOV SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
SFA-MOV-EQ-AP1S2 PUMP 1 MOV SEISMIC FRAGILITY (0.3) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
SFA-MOV-EQ-AP1S3 PUMP 1 MOV SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
SFA-MOV-EQ-AP2S1 PUMP 2 MOV SEISMIC FRAGILITY (0.1) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS1 template event.	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
SFA-MOV-EQ-AP2S2 PUMP 2 MOV SEISMIC FRAGILITY (0.31) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS2 template event.	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
SFA-MOV-EQ-AP2S3 PUMP 2 MOV SEISMIC FRAGILITY (0.6) *Note:* This event uses the TEMPEV-MOV-EQ-SEIS3 template event.	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
SFA-MOV-RP-P1 SULFURIC ACID PUMP 1 MOV FAILS	**5.00E-07**	Lognormal(EF=10.0) 95th = 1.88E-6

Note:

SFA-MOV-RP-P2	**5.00E-07**	Lognormal(EF=10.0)
SULFURIC ACID PUMP 2 MOV FAILS		95th = 1.88E-6
Note:		
SFA-PIP-EQ-SAS1	**1.26E-09**	Constrained non-informative
SULFURIC ACID PIPING SIESMIC FRAGILITY (0.1)		95th = 5.48E-9
Note: This event uses the TEMPEV-PIP-EQ-SEIS1 template event.		
SFA-PIP-EQ-SAS2	**1.04E-05**	Constrained non-informative
SULFURIC ACID PIPING SIESMIC FRAGILITY (0.3)		95th = 4.52E-5
Note: This event uses the TEMPEV-PIP-EQ-SEIS2 template event.		
SFA-PIP-EQ-SAS3	**1.65E-03**	Constrained non-informative
SULFURIC ACID PIPING SIESMIC FRAGILITY (0.6)		95th = 7.17E-3
Note: This event uses the TEMPEV-PIP-EQ-SEIS3 template event.		
SFA-PIP-RP-ACID	**7.20E-07**	Lognormal(EF=10.0)
SULFURIC ACID DAMAGES PIPING		95th = 2.70E-6
Note:		
SFA-TNK-EQ-STS1	**9.87E-08**	Constrained non-informative
ACID TANK SEISMIC FRAGILITY (0.1)		95th = 4.29E-7
Note: This event uses the TEMPEV-TNK-EQ-SEIS1 template event.		
SFA-TNK-EQ-STS2	**1.61E-03**	Constrained non-informative
ACID TANK SEISMIC FRAGILITY (0.3)		95th = 7.00E-3
Note: This event uses the TEMPEV-TNK-EQ-SEIS2 template event.		
SFA-TNK-EQ-STS3	**1.15E-01**	Constrained non-informative
ACID TANK SEISMIC FRAGILITY (0.6)		95th = 4.70E-1
Note: This event uses the TEMPEV-TNK-EQ-SEIS3 template event.		
SFA-TNK-FC-TNK	**1.00E-07**	Lognormal(EF=10.0)
SULFURIC ACID STORAGE TANK RUPTURE		95th = 3.75E-7
Note:		
SFA-TNK-RP-TNK	**1.00E-07**	Lognormal(EF=10.0)
SULFURIC ACID STORAGE TANK OVERFILL		95th = 3.75E-7
Note:		
SFA-XHE-XA-BUR	**1.00E-03**	Lognormal(EF=10.0)
OPERATOR ACTION CAUSES BURNER EXPLOSION		95th = 3.75E-3
Note:		
TEMPEV-ACC-EQ-SEIS1	**1.45E-12**	Constrained non-informative
TEMPLATE EVENT SEISMIC FRAGILITY FOR ACCUMULATOR (0.1g)		95th = 6.30E-12
Note:		
TEMPEV-ACC-EQ-SEIS2	**1.11E-06**	Constrained non-informative
TEMPLATE EVENT SEISMIC FRAGILITY FOR ACCUMULATOR (0.3g)		95th = 4.84E-6
Note:		
TEMPEV-ACC-EQ-SEIS3	**1.43E-03**	Constrained non-informative
TEMPLATE EVENT SEISMIC FRAGILITY FOR ACCUMULATOR (0.6g)		95th = 6.22E-3
Note:		
TEMPEV-FAN-EQ-SEIS1	**6.06E-11**	Constrained non-informative
TEMPLATE EVENT SEISMIC FRAGILITY FOR AIR HANDLING UNIT (0.1g)		95th = 2.64E-10
Note:		

TEMPEV-FAN-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR AIR HANDLING UNIT (0.3g) *Note:*	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
TEMPEV-FAN-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR AIR HANDLING UNIT (0.6g) *Note:*	**2.98E-03**	Constrained non-informative 95th = 1.29E-2
TEMPEV-HTX-EQ-SEIS1 TEMPLATE EVENT SEISMIC FRAGILITY FOR HTX (0.1g) *Note:*	**8.44E-11**	Constrained non-informative 95th = 3.67E-10
TEMPEV-HTX-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR HTX (0.3g) *Note:*	**1.81E-05**	Constrained non-informative 95th = 7.87E-5
TEMPEV-HTX-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR HTX (0.6g) *Note:*	**8.47E-03**	Constrained non-informative 95th = 3.67E-2
TEMPEV-MDP-EQ-SEIS1 TEMPLATE EVENT SEISMIC FRAGILITY FOR MOTOR DRIVEN PUMP (0.1g) *Note:*	**4.05E-11**	Constrained non-informative 95th = 1.76E-10
TEMPEV-MDP-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR MOTOR DRIVEN PUMP (0.3g) *Note:*	**1.11E-05**	Constrained non-informative 95th = 4.83E-5
TEMPEV-MDP-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR MOTOR DRIVEN PUMP (0.6g) *Note:*	**6.23E-03**	Constrained non-informative 95th = 2.70E-2
TEMPEV-MOV-EQ-SEIS1 TEMPLATE EVENT SEISMIC FRAGILITY FOR MOV (0.1g) *Note:*	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
TEMPEV-MOV-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR MOV (0.3g) *Note:*	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
TEMPEV-MOV-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR MOV (0.6g) *Note:*	**1.65E-03**	Constrained non-informative 95th = 7.17E-3
TEMPEV-PIP-EQ-SEIS1 TEMPLATE EVENT SEISMIC FRAGILITY FOR PIPING (0.1g) *Note:*	**1.26E-09**	Constrained non-informative 95th = 5.48E-9
TEMPEV-PIP-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR PIPING (0.3g) *Note:*	**1.04E-05**	Constrained non-informative 95th = 4.52E-5
TEMPEV-PIP-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR PIPING (0.6g) *Note:*	**1.65E-03**	Constrained non-informative 95th = 7.17E-3

TEMPEV-TNK-EQ-SEIS1 TEMPLATE EVENT SEISMIC FRAGILITY FOR STORAGE TANK (0.1g) *Note:*	**9.87E-08**	Constrained non-informative 95th = 4.29E-7
TEMPEV-TNK-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR STORAGE TANK (0.3g) *Note:*	**1.61E-03**	Constrained non-informative 95th = 7.00E-3
TEMPEV-TNK-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR STORAGE TANK (0.6g) *Note:*	**1.15E-01**	Constrained non-informative 95th = 4.70E-1
TEMPEV-VES-EQ-SEIS1 TEMPLATE EVENT SEISMIC FRAGILITY FOR PRESSURIZED VESSEL (0.1g) *Note:*	**6.06E-11**	Constrained non-informative 95th = 2.64E-10
TEMPEV-VES-EQ-SEIS2 TEMPLATE EVENT SEISMIC FRAGILITY FOR PRESSURIZED VESSEL (0.3g) *Note:*	**6.59E-06**	Constrained non-informative 95th = 2.87E-5
TEMPEV-VES-EQ-SEIS3 TEMPLATE EVENT SEISMIC FRAGILITY FOR PRESSURIZED VESSEL (0.6g) *Note:*	**2.98E-03**	Constrained non-informative 95th = 1.29E-2